RADIATION AND PROTECTION

U. Ya. Margulis

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16. Abstract

This book describes the interaction of ionizing radiation with matter, the action of radiation on living organisms, and the protective measures necessary in working with radioactive materials and with sources of ionizing radiation. Information regarding atomic structure and nature of the interaction of radiation with matter are presented consisely.

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ANNOTATION

This book describes the interaction of ionizing radiation with matter, the action of radiation on living organisms, and the protective measures necessary in working with radioactive materials and with sources of ionizing radiation. Information regarding atomic structure and nature of the interaction of radiation with matter are presented concisely.

In view of the appearance of "Standards of Radiation Protection SRP-69" and "Principal Sanitary Rules", a greater share of the attention is devoted to the treatment of the more important aspects of these documents in the third edition of this book (2nd edition in 1969).

This book is designed for readers with a high school background in physics, working with ionizing radiation.

FOREWORD TO THE THIRD EDITION

In view of the rapid development of atomic energy and the ever- /3*
increasing scale of application of radioactive substances and sources of ionizing radiation in various sectors of the national economy, interest is growing in radiation protection not only on the part of specialists, but also on the part of the general population. This book presents, in an accessible manner, problem concerning the action of ionizing radiation on living organisms, as well as protective measures which must be taken in order to eliminate adverse effects of radiation.

After publication of the second edition of this book, "Standards of Radiation Protection SRP-69" and "Principle Sanitary Rules Governing the Use of Radioactive Substances and Other Sources of Ionizing Radiation PSR-72" were implemented, superseding "Maximum Permissible Levels MPL-60" and "Sanitary Rules Governing the Use of Radioactive Substances and Sources of Ionizing Radiation SR-333-60". As a result, we felt it necessary to devote greater attention to the treatment of the more important aspects of these new documents, and to substantiate the changes which were incorporated in the existing standards.

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*Numbers in the margin indicate pagination in the original foreign text.

Along with this, the new edition examines in greater detail the biological action of ionizing radiation, action of radiation on the human organism, the principles of dosimetric control, and a description of new dosimetric instrumentation. Furthermore, the third edition includes new material dealing with the biological consequences of irradiation, the rationale leading to the establishment of permissible levels of external irradiation, the regulation of the internal use of radioactive substances, etc.

In conclusion, I consider it my pleasant obligation to express my sincere appreciation to Doctor of Medicine I. K. Dibobes, who carefully read the manuscript and gave a number of valuable suggestions.

FOREWORD TO THE SECOND EDITION [1969]

Eight years have elapsed since the publication of the first edition of the book "Protection From the Action of Penetrating Radiation", published by Gosatomizdat in 1961. During this time, atomic energy has found ever-widening applications in practically all areas of science and technology. Atomic technology has undergone continued growth. These factors were taken into account in the present edition of this book, whose aim it is to acquaint a wide group of readers with modern concepts of the action of radiation on living organisms, permissible levels of radiation, and methods which will assure radiation safety in working with radioactive substances and sources of ionizing radiation. In particular, Chapter 1 includes material dealing with nuclear fission, and principles of operation of nuclear reactors. /4

Chapters 3 and 4 present new material on the biological effects of radiation. In accordance with the latest recommendations of the International Commission on Radiation Protection, the principles underlying the establishment of permissible levels of radiation are given.

Chapter 4 presents those changes that were incorporated in the maximum permissible levels and maximum permissible concentrations, in accordance with the recommendations of the Permanent Commission

on the Utilization of Atomic Energy for peaceful purposes of the Council of Economic Mutual Assistance.

Chapter 5 includes a description of new dosimetric instrumentation, which has found practical applications. Some changes dealing with systems providing radiation protection are covered in Chapter 6.

In view of the fact that, in comparison with the first edition, the book has been substantially modified, we found it appropriate to give it a new name "Radiation and Protection", which is in better agreement with its contents.

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INTRODUCTION

Just as the preceding century was the age of steam and electricity, the twentieth century became the age of atomic energy, which plays a revolutionary role and contributes enormously to scientific and technological progress. At the present time, atomic energy is being set up to supply the national economy of our country. At present, it is very likely that there is not a single branch of science or technology which does not, in some manner, use the energy released by nuclear transformations. Radioactive isotopes are successfully used in medicine, in the treatment of malignant neoplasms and other serious illnesses, as well as in the diagnosis of a number of illnesses and in the investigation of the functional condition of the organism. In industry, the quality control of products, the control of technological processes, the determination of the structure of alloys, verification of material separation methods, etc., are all done with the help of radioactive isotopes. /5

The application of radioactive isotopes has significantly increased the possibility of experimental investigation of wear of machine components and assemblies, the behavior of lubricants in friction areas of machines, and in the analysis of liquid and gaseous media.

Radioactive isotopes are being ever more widely used in the control of agricultural pests, in the disinfestation of grain, in the

introduction of new varieties of grain, in the acceleration of chemical reactions, the development of new polymers, the investigation of catalytic processes, cryogenic sterilization, binding and medicinal preparations, in geological surveys, and in other sectors of the national economy.

The application of radioactive isotopes made it possible not only to improve significantly scientific investigations in many areas of knowledge, but also to equip scientists with new methods of actively influencing nature. /6

The use of radioactive tracer isotopes made possible the study of new principles and the attainment of important new discoveries in biology, chemistry, metallurgy, and even archeology.

The Soviet people are proud that the genius of our scientists was the first in the world to apply the energy hidden in the atom to turn the turbines of a power plant and the screw propellers of the atomic icebreaker "Lenin". A new atomic icebreaker "Arktik" is being constructed at the present time.

As is well known, the initial application of the idea of atomic energy utilization in the production of electrical energy was the operation on June 27, 1954, of the first atomic electrical power plant with a capacity of 5000 kilowatts. The technical initiative of the Soviet Union in the application for peaceful purposes of the achievements of nuclear science has found general recognition.

At the present time, nuclear technology is developing rapidly. It is sufficient to state that up to January 1, 1972, 100 atomic electric power-generating plants with a capacity of 30 million kilowatts [1] were put into operation in a number of countries. Thus, the capacity of atomic power generating plants increased approximately by 6000 times in less than 20 years. A comparable increase in the capacity of conventional power plants would have required approximately 100 years. This clearly testifies to the fact that further

development of the national economy is impossible without progress in the area of nuclear technology.

We are now on the threshold of the application of atomic energy in a new area of human endeavor: the utilization of nuclear-powered devices in space. Space vehicles are already using atomic electrical generating plants and isotope sources of electrical and thermal energy. The time is not far removed when the wide spaces of the universe will be traversed by space vehicles using nuclear propulsion.

Naturally, with such widespread utilization of atomic energy, the number of people who may be exposed to the action of ionizing radiation increases yearly. Therefore, just as in the large scale use of any other form of energy (thermal, mechanical, electrical, or chemical), the correct organization of work and the implementation of protective and prophylactic measures assume a great importance.

It is known that the action of a large amount of thermal, mechanical, electrical energy or of a number of chemical agents can not only disturb the normal life activity of the human organism, but also at times bring about its death. The same occurs for certain doses of ionizing radiation which bring about adverse effects on the human organism. /7

It is important to note that radiation has properties which make it necessary to take special safety measures.

It is known that our sensory organs are very sensitive to changes in the photo, mechanical, or thermal environment. This helps man to orient himself in a given situation and to take corresponding precautionary measures. At the same time, our senses are not adapted to the detection of ionizing radiation. As a result, we cannot make a judgment about the presence and level of radiation without special instruments and, consequently, also about the impending danger. As a result, the growth of nuclear energy stimulated the development of a branch of knowledge such as radiation safety which is devoted to the development of an effective system of control of the radiation

levels, to the determination of permissible levels of irradiation, to the development of collective and individual methods of radiation protection, and to problems of organization of work under conditions of irradiation.

In order to eliminate the adverse effects of ionizing radiation, a special discipline and a special organization of work with radioactive substances and sources of ionizing radiation are required. At the same time, the dangers should not be overexaggerated, as some have a tendency to do, believing that man is not protected from the adverse effects of radiation under any working conditions.

Due to the enormous attention devoted to radiation safety by society in all undertakings and at all institutions using radioactive substances and sources of ionizing radiation, effective methods of protection have been created. As a result, the number of unfortunate accidents and occupationally related illnesses is significantly lower than in any other branch of industry. It is sufficient to indicate that the great majority of workers are exposed to doses less than the maximum permissible ones [2].

This book attempts to describe the nature of the interaction of /8
ionizing radiation and the protective measures which make work with radioactive substances as safe as, and in a number of cases safer than, any other. For the sake of completeness and unity of presentation, it seems appropriate to cover briefly the atomic structure and the nature of the interaction of radiation with matter. This will give the reader who is not a specialist in radiation safety the opportunity to review the general concepts of atomic physics and dosimetry without reference to other sources.

CHAPTER 1

PROPERTIES OF IONIZING RADIATIONS

Atomic Structure and Radioactivity

Discovered by D. I. Mendeleyev, the periodic law of the variation of chemical properties of elements with increasing atomic weight, logically forces us to admit that atoms have a complex structure. It is difficult to imagine, otherwise, the existence of a regular variation in their properties if we consider them to be indivisible, i.e., not having an internal structure. /9

The discovery by Henri Becquerel of an invisible radiation emitted by uranium and its compounds, as well as the classical work by Marie Sklodowska Curie and Pierre Curie established the nature of these invisible rays, put an end to the idea of an indivisible atom, and marked the beginning of man's penetration into the secrets of atomic structure.

Experiments by Rutherford and his students indisputably showed that the atom, for ages considered to be the smallest indivisible particle of the universe, had a complex structure. It is made up of a positively charged nucleus, containing 99.95% of the atomic mass, with electrons rotating about this nucleus. The cross-sectional dimensions of the nucleus are approximately ten thousand times smaller than those of the atom.

Further experiments have shown that the atomic nucleus also has a complex structure and contains protons (hydrogen nuclei) and neutrons.* The proton has a unit positive charge**, while its atomic mass is approximately equal to one***. The neutron is a neutral particle whose mass is approximately equal to that of the proton. /10

The number of protons in a nucleus is equal to the atomic number of the element in Mendeleyev's table, while the number of neutrons is equal to the difference between the mass number and the atomic number. The mass number is the integral number closest to the atomic mass of the isotope of a given chemical element. The mass number is equal to the total number of particles (protons and neutrons) in the nucleus. For example, the atomic masses of potassium and uranium are 39.1 and 238.03, respectively. Therefore, the corresponding mass numbers are 39 and 238.

The number of protons in a nucleus determines its charge; consequently, the charge on a nucleus is equal to the atomic number of the element. Since the atom is neutral in its normal state, the number of electrons, orbiting around the atom, is also equal to the atomic number in Mendeleyev's table.

Electrons are located in shells around the nucleus. Furthermore, each shell may have a certain specified number of electrons. For example, the shell closest to the nucleus (K-shell) may have, at

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*The proton-neutron hypothesis of nuclear structure was first proposed in 1932 by Soviet physicists Ye. G. Gapon and D. D. Ivanenko.

**In atomic physics, the unit of charge is given by the absolute magnitude of the charge on the electron, equal to $4.8 \cdot 10^{-10}$ electrostatic units (CGSE).

***Atomic mass is a number which indicates the number of times that the mass of a given nuclear particle or atom is greater than 1/12 of the mass of a carbon atom (or atomic mass unit). The atomic mass unit is equal to $1.66 \cdot 10^{-24}$ grams. The atomic mass of a proton is equal to 1.00813; that of a neutron is 1.00898. The mass of an electron is approximately 1840 times lighter than that of a proton.

most, two electrons; the next shell (L-shell) — eight electrons; the third shell (M-shell) — 18 electrons, and so on.

A restructuring of the electron shells only occurs during all chemical reactions; furthermore, only outermost shells are affected, where electrons are least tightly bound to the nucleus. The nucleus does not take part in chemical reactions. Consequently, similar chemical properties are common only to chemical elements having the same number of electrons in the outermost shell. Thus, the periodic variation in the chemical properties of elements with increasing atomic mass depends on the periodic structure of electron shells.

Elements having similar chemical properties, but different mass number (or correspondingly different atomic masses), are called isotopes. It is clear that nuclei of isotopes of one and the same element are composed of the same number of protons and different number of neutrons. /11

Nuclear forces of attraction act between particles constituting the nucleus, i.e., between protons and protons, neutrons and neutrons, and protons and neutrons. The special characteristic of these forces is that they are exceedingly strong at distances on the order of the size of the nucleus (10^{-13} cm), and decrease rapidly with increasing interparticle distances. In addition to nuclear forces of attraction, Coulomb repulsive forces exist between similarly charged nuclear particles — protons. For most chemical elements, the attractive nuclear forces overcome the repulsive nature of the Coulomb force, a factor which determines the nuclear stability of these elements.

However, the attractive nuclear forces are not able to compensate for the Coulomb force of repulsion in heavy elements, whose nuclei are made up by a great number of particles. In this case, processes of internal reorganization of the nuclei are initiated. These processes lead to a spontaneous change of the nuclei from a less stable to a more stable state. This phenomenon, discovered by Henri Becquerel and investigated by Marie Sklodowska Curie and Pierre Curie, was called radioactivity.

At the present time, more than 40 naturally radioactive elements, which are isotopes of elements with atomic numbers greater than 82, are known. Some lighter, naturally occurring isotopes are radioactive as well. For example, H^3 (tritium), K^{40}, C^{14}, Rb^{87} (the superscript indicates the mass number).

More than 1000 different artificially radioactive isotopes have been produced presently in nuclear reactors and in accelerators. Many of these isotopes are widely used in various areas of science and technology, medicine, and agriculture.

In the process of radioactive decay, the atomic nuclei emit either an α-particle or a β-particle. In addition, as a rule, all nuclei of a given radioactive isotope emit particles of the same type.

Alpha particles, a flux of helium nuclei, are composed of two proton and two neutrons. Thus an α-particle has two units of positive charge, while its atomic mass is equal to 4. All nuclei of a given radioactive isotope emit α-particles of the same energy*. The energies of α-particles emitted by presently known radioactive isotopes lie between 3 - 9 MeV**. /12

Beta particles — flux of electrons or positrons***.

The first refers to electron β^--decay, the second — to positron β^+-decay.

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*We will not consider here long and short range alpha particles, which are rather uncommon.

**Electronvolt (eV) — unit of energy used in atomic physics. It is equal to the kinetic energy acquired by an electron accelerated through a potential difference of one volt; 1 eV = $1.6 \cdot 10^{-12}$ ergs; 1 keV (kiloelectron volt) = 10^3 eV = $1.6 \cdot 10^{-9}$ ergs; 1 MeV (megaelectronvolt) = 10^6 eV = $1.6 \cdot 10^{-6}$ ergs.

***Positron — particle having the same mass as the electron with unit positive charge.

In contrast with alpha decay, different nuclei of a given radioactive isotope emit in β-decay, beta particles with various energies. At the same time, a maximum beta particle energy is characteristic for a given radioactive isotope. Thus, the energy spectrum of β-particles is continuous (Figure 1). The continuous nature of the β-spectrum is due to the fact that two particles are emitted from the nucleus during β-decay: an electron (or positron) and a neutrino*. The excess energy of the nucleus of a given radioactive isotope is shared differently by the two particles. The maximum energies of the β-spectrum produced by presently known radioactive isotopes range from several tens of kiloelectronvolts to 3.-3.5 MeV.

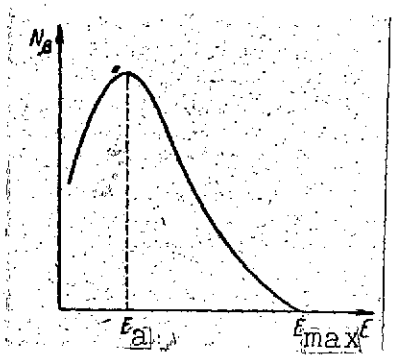

Figure 1. Distribution of β-particles according to energy in a simple β-spectrum: N_β — number of β-particles; E — their energy

Let us consider what occurs in the nucleus during the process of radioactive decay. The number of protons and neutrons in the nucleus decreases by two during the emission of α-particles. Thus, in α-decay, a new element is formed, with an atomic number smaller by two units, and a mass number smaller by four units. For example, a new element is formed — radon Ra_{86}^{222} during α-decay of radium Ra_{88}^{226}**. /13

The emission of electrons or positrons from the nucleus during β-decay depends on the fact that under certain conditions, one of the neutrons of the nucleus is transformed into a proton, or one of the protons of the nucleus — into a neutron:

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*Neutrino — neutral particle with rest mass equal to zero; it has a very great penetrating power.

**The subscript indicates the atomic number; the superscript — the mass number.

$$n_0^1 \rightarrow p_1^1 + e^- + \nu;$$

neutron proton electron neutrino

$$p_1^1 \rightarrow n_0^1 + e^+ + \nu.$$

proton neutron positron neutrino

An electron is emitted from the nucleus in the first reaction, i.e., an electron β^--decay takes place. The total number of particles in the newly formed nucleus remains unchanged, with the number of protons increasing by one. Thus, in electron β^--decay, a new nucleus is formed with the same mass number and an atomic number greater by one. For example, the naturally radioactive isotope Bi_{83}^{209} in the process of β^--decay is transformed into an isotope with an atomic number of 84 and the same mass number, i.e., into Po_{84}^{209}.

In positron β^+-decay, the proton is transformed into a neutron with the emission of a positron. Therefore, the newly formed nucleus will have an atomic number smaller by one and the same mass number.

In some cases, radioactive transformations occur without the emission of particles from the nucleus, but with the capture of an orbital electron by the radioactive nucleus. As a result, one of the protons of the nucleus is transformed into a neutron. This process is known as K-capture, since the capture of a K-shell electron takes place. It is clear that during K-capture, the newly formed nucleus will have an atomic number smaller by one and the same mass number, just as it does during positron β^+-decay.

In some cases, the daughter nucleus, formed as a result of radioactive decay, may be in an excited state. The transition of the nucleus from the excited to a ground state is accompanied by the emission of gamma rays. The energy of gamma photons emitted during the process of radioactive decay ranges from a few kiloelectronvolts to 3 - 4 MeV. /14

Gamma photons, just as radiowaves, visible light, ultraviolet and infrared rays, as well as x-rays, are electromagnetic waves which propagate in a vacuum with a constant velocity equal to 300,000 km/s ($3 \cdot 10^{10}$ cm/s). The various forms of radiation, having the same nature, differ in the methods of production, as well as in their properties (wavelengths or energy). For example, x-ray radiation or, as it is known, bremsstrahlen, is produced as a result of slowing down of fast electrons.

Visible light, infrared and ultraviolet rays, and characteristic x-rays are emitted in various atomic and molecular transitions from excited to ground states.

Gamma quanta are emissions of nuclear origin, produced during radioactive decay or nuclear reactions by the transition of a nucleus from one energy state to another. According to current understanding, radiation is one form of mass. In the process of interaction with matter, radiation exhibits wave as well as particle properties. Thus, radiation can be characterized by a specific wavelength, or it can be considered to be a beam of uncharged particles, photons or quanta, which have a definite mass and energy. The photon energy

$$E = h\nu \tag{1}$$

where $h = 6.623 \cdot 10^{-10}$ ergs $\cdot$ sec is the quantum of action, or Planck's constant, and ν is the frequency of radiation. It is known that:

$$\nu = \frac{c}{\lambda} \tag{2}$$

where λ is the wavelength, and $c = 3 \cdot 10^{10}$ cm/sec is the velocity of propagation of radiation in a vacuum.

Radioactive Decay Law

The special characteristic of the phenomenon of radioactive decay consists in the fact that not all nuclei of a given radioactive element decay simultaneously, but that a definite fraction of the

nuclei decay per unit time. The fraction of decaying nuclei differs /15
for different radioactive elements. For example, $1.38 \cdot 10^{-11}$ of the total number of radium atoms decay per second, while $2.1 \cdot 10^{-6}$ of the total number of radon atoms decay in the same time. The fraction of atoms decaying per unit time is known as the decay constant λ. The decay constant is given in units of sec^{-1}, day^{-1}, $year^{-1}$, etc.

It has been shown that the decrease of radioactive atoms with time is given by:

$$N_t = N_0 e^{-\lambda t} \qquad (3)$$

where N_t is the number of remaining radioactive atoms after a time t; N_0 is the number of atoms of the radioactive element at some initial time; e is the base of natural logarithms (e = 2.718); λ is the decay constant. Figure 2 presents an exponential curve and illustrates the nature of the decreasing number of radioactive atoms, given by Equation (3). We can see that after some time T, the number of radioactive atoms decreases by a factor of 2. The time during which the number of radioactive atoms decrease by a factor of two is known as the half-life T. The following relationship exists between the half-life and the decay constant:

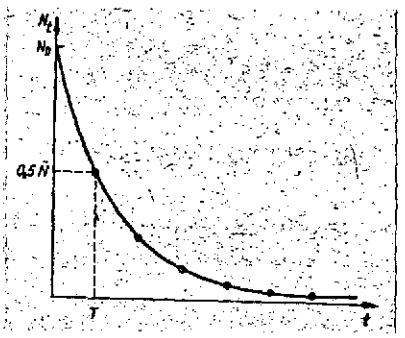

Figure 2. Change in the number of radioactive atoms N_t with time t

$$\lambda = \frac{\ln 2}{T} = \frac{0{,}693}{T}; \qquad (4)$$

after substitution in Equation (3), we obtain:

$$N_t = N_0 e^{-\frac{0{,}693}{T} t} \qquad (5)$$

The half life of different radioisotopes varies over a very large range — from billions of years to microseconds. For example,

the half life of uranium U_{92}^{238} is 4.5 billion years, that of radium Ra_{88}^{226} — 1622 years, that of radon Rn_{86}^{222} — 3.83 days, etc. /16

During each radioactive decay, approximately a million times more energy is liberated than during such violent chemical reactions as combustion (during the combination of carbon and oxygen atoms). However, the practical utilization of the enormous energy of radioactive decay presents a difficult problem, since only a predetermined fraction of the atoms of a given quantity of the substance decay per unit time. The process takes place slowly, and cannot be accelerated in any manner. In combustion, as opposed to radioactive decay, nearly all the atoms of a given amount of material react within a short period of time.

Units of Activity

In working with radioactive substances, the pertinent parameter is not their mass, but rather the number of emitted particles or the quantity proportional to it — the number of decaying nuclei. The concept of "activity" was introduced to quantize the characteristics of radioactive substances. The activity of a radioactive isotope is determined by the number of atoms decaying per unit time.

The unit of activity is the Curie, which corresponds to $3.7 \cdot 10^{10}$ disintegrations per second. The derived units are millicuries (mCi), microcuries (μCi), nanocuries (nCi), and picocuries (pCi):

$$1\ mCi = 10^{-3}\ Ci = 3.7 \cdot 10^{7}\ dis/s;$$

$$1\ \mu Ci = 10^{-6}\ Ci = 3.7 \cdot 10^{4}\ dis/s;$$

$$1\ nCi = 10^{-9}\ Ci = 37\ dis/s;$$

$$1\ pCi = 10^{-12}\ Ci = 3.7 \cdot 10^{-2}\ dis/s.$$

It should be noted that approximately $3.7 \cdot 10^{10}$ decays occur per second in one gram of purified radium, with daughter products removed. Thus, the activity of one gram of pure radium is close to

one curie. Therefore, the activity of radium is sometimes given in grams, since only in this case does the unit of mass correspond to a unit of activity.

Since the number of radioactive atoms decreases with time, the activity also decreases. It is apparent that this decrease occurs /17
according to a law similar to (3), that is:

$$C_t = C_0 e^{-\lambda t} = C_0 e^{-\frac{0{,}693}{T} t} \quad (4)$$

where C_t is the activity after a time t, C_0 is the activity at some initial time; λ and T are the corresponding decay constant and half-life.

Nuclear Reactions and Artificial Radioactivity

The discovery of the phenomenon of radioactivity destroyed the myth of indivisibility of the atom, and indisputably showed the possibility of the transmutation of one element into another.

Naturally, scientists could not be satisfied only with the observation of those nuclear transformations which occur in nature. Their efforts were directed at finding methods of penetrating into the nucleus of the atom and producing its transformation. This would bring about not only the fulfillment of the age-old wish of the alchemists to transform one element into another, but would also give rise to the possibility of obtaining an inexhaustible source of energy.

The great physicist of our time, Ernest Rutherford, was the first to penetrate the secrets of the atom. He used alpha particles emitted by naturally radioactive elements as projectiles to bombard the nucleus. These particles, having enormous velocities (on the order of 20,000 km/sec), had a finite probability of penetrating into the nucleus, after overcoming the Coulomb force of repulsion between similarly charged alpha particles and the nucleus. The hypothesis of the scientist was brilliantly confirmed. Rutherford discovered

that in the bombardment of the nitrogen N_7^{13} nuclei by alpha particles (helium He_2^4 nuclei), a proton (hydrogen nucleus H_1^1) is emitted, and an oxygen O_8^{16} nucleus is formed:

$$N_7^{14} + He_2^4 \rightarrow O_8^{17} + H_1^1. \quad (6)$$

The formation of new elements also occurs in the bombardment of other elements by alpha particles. For example, boron nuclei are formed from lithium nuclei:

$$Li_3^7 + He_2^4 \rightarrow B_5^{10} + n_0^1, \quad (7)$$

silicon — from aluminum, etc.

The mechanism of a nuclear reaction can be described in the following manner. On penetrating into the nucleus, the particle imparts its energy to all the protons and neutrons of the nucleus. A compound nucleus is formed in an excited state, and a continuous exchange of energy between the nucleons takes place. If, at any given moment, one or more of the constituent particles (protons, neutrons) accumulate enough energy to overcome nuclear forces of attraction, then one or more particles (proton, neutron, alpha particles) will be ejected from the nucleus. As a result, a nucleus will be formed with a number of protons and neutrons different from the parent one. The lifetime of the compound nucleus is infinitesimally small, and is approximately 10^{-13} seconds. /18

For example, the penetration of an alpha particle into an aluminum nucleus, which is made up of 13 protons and 14 neutrons, forms a compound nucleus with an atomic number of 15 and a mass number of 31, i.e., an isotope of phosphorus. The compound nucleus of phosphorus emits a proton and forms a silicon nucleus with atomic number of 14 and mass number of 30:

$$Al_{13}^{27} + He_2^4 \rightarrow {}^*P_{15}^{31} \rightarrow Si_{14}^{30} + H_1^1. \quad (8)$$

It may happen that none of the particles will accumulate enough energy to overcome the binding force of the nucleus. In this case, the excess energy, imparted to the nucleus by the particle, will be emitted in the form of a gamma photon. For example,

$$C_6^{12} + H_1^1 \rightarrow N_7^{13} + \gamma. \quad (9)$$

At the present time, modern accelerators can accelerate nuclear particles (protons, neutrons, deuterons, and alpha particles) up to enormous energies. This makes it possible to obtain a large number of of nuclear reactions which produce not only isotopes of known chemical elements, but also produce new previously unknown chemical elements with atomic numbers greater than 92, the so-called transuranium elements.

Stable as well as unstable, i.e., artificial radioactive, isotopes may be formed during nuclear reactions.

Artificial radioisotopes, as a rule, undergo β^+- or β^--decay. Among some artificial radioisotopes, β-decay is accompanied by the emission of gamma photons, provided that the daughter nucleus is formed in an excited state. /19

Artificial radioisotopes may be produced by the bombardment of nuclei of stable isotopes with α-particles, protons, deuterons, and neutrons. The most effective method of artificial radioisotope production is the bombardment by neutrons, since these particles are uncharged and do not require large energies to penetrate into the nucleus, as is the case with an α-particle, proton, or deuteron. Neutrons of very low energies, corresponding to thermal vibrations of atoms and molecules, the so-called thermal neutrons (energy 0.025 eV) easily penetrate into atomic nuclei, bringing about a regrouping with the formation of artificial radioisotopes of the same element. For example,

$$Co_{27}^{59} + n_0^1 \rightarrow Co_{27}^{60} + \gamma. \quad (10)$$

The product isotope Co_{27}^{60} is an artificial radioisotope. By β^--decay, it is transformed into a stable isotope of nickel (Ni_{28}^{60}). In the decay process of the Co_{27}^{60} nucleus, two gamma photons are emitted with energies equal to 1.17 and 1.33 MeV.

Artificial radioactivity was discovered in 1934 by the well known French physicists Frederick and Irene Joliot-Curie, for which they were awarded the Nobel prize.

It is difficult to overestimate the significance of this discovery. At the present time, more than 1000 different artificial radioisotopes have been produced. Many of these have received wide applications, in practically all areas of science, technology, medicine, agriculture, etc. [3].

Nuclear reactions can proceed with the emission, as well as with the absorption, of energy. The amount of energy liberated during each nuclear reaction is a hundred thousand times greater than that evolved during combustion. However, as it appeared initially, a number of insuperable obstacles blocked the way to the practical utilization of nuclear energy. The essence of these consisted in the following: exoergic chemical reactions are self-sustaining. This implies that only a small amount of energy is required to initiate a /20
chemical reaction. In this case, some number of atoms (for example, carbon and oxygen atoms in the combustion process) enter into a reaction, and the evolved energy further stimulates the ongoing reaction. As a result, the number of atoms entering into the reaction increases exponentially. A chain reaction is initiated, and a quantity of energy is evolved which significantly exceeds the energy required to initiate the given reaction.

A completely different situation exists with nuclear reactions. It is true that a nuclear reaction occurs when an α-particle penetrates into an aluminum nucleus, with the production of a silicon nucleus and the evolution of energy equal to 2.4 MeV, approximately

seven hundred thousand times greater than the energy evolved during the combustion reaction (combination of one carbon and oxygen atom). However, the evolved energy cannot stimulate additional reactions, as it does, for example, in combustion.

In order to bring about the formation of a new silicon nucleus and the evolution of a definite amount of energy, a new α-particle capable of penetrating into another aluminum nucleus is required.

In view of the fact that the size of the nucleus is many times smaller than that of the atom, the probability of an α-particle or proton penetrating into a nucleus is infinitesimally small. Approximately one of every 100,000 α-particles or protons can penetrate into the nucleus. Furthermore, in order to penetrate into the nucleus, the α-particle or proton has to overcome the Coulomb repulsion, which increases as the particle approaches the nucleus. Therefore, the α-particles or protons suitable for nuclear bombardment must be first accelerated to enormous energies. As a result, the energy used to produce a nuclear reaction exceeds the evolved energy and, most importantly, does not bring about a self-sustaining nuclear chain reaction.

During the first forty years of this century, in spite of the efforts of scientists all over the world, the human race was as far from the practical utilization of atomic energy as it was at the time of Rutherford's production of the first nuclear reaction.

However, early in 1939, a remarkable discovery produced a ray of hope.

Nuclear Fission

After the discovery of the neutron in 1932 and of artificial radioactivity in 1934, scientists became interested in "modern alchemy", i.e., the production of new radioactive elements by the action of neutrons. /21

Fermi, young at that time, attempted to obtain the new unknown 93^{rd} element by neutron bombardment of uranium, the 92^{nd} element in Mendeleyev's table. However, as a result of neutron capture by uranium nuclei, not just one, but at least ten artificial radioactive elements were found.

Nature presented man with a new problem. It can be considered that a new stage in the development of nuclear physics was begun at that moment. The possibility of utilizing the energy hidden in the depths of the atom became a reality.

The explanation of the new phenomenon was given by Frederick Joliot-Curie and Lise Meitner. They showed that in the process of neutron bombardment of uranium, a new type of nuclear reaction takes place — the splitting of the uranium nucleus into two approximately equal parts (fragments). The energy evolved during this reaction is approximately 200 MeV, i.e., tens of time greater than that associated with the then commonly known nuclear reactions.

The theory of uranium fission was worked out simultaneously and independently by the Soviet scientist Frenkel and the Dutch scientist Bohr.

The special feature of the uranium fission reaction consists in that, in addition to the two fragments, two or three neutrons are formed in each fission which could bring about the fission of other nuclei. During each of these processes, new neutrons are liberated which, in turn, bring about the fission of other nuclei (Figure 3). Thus, one neutron can initiate a fission chain reaction, with the number of nuclei undergoing fission increasing exponentially. The uranium fission reaction develops as a chain reaction. For example, all nuclei contained in 1 kg of uranium (approximately $3 \cdot 10^{24}$ nuclei) will undergo fission in a fraction of a second. The energy evolved in this is equal to the energy released by an explosion of 20,000 tons of TNT, or by the combustion of 2.5 thousand tons of coal.

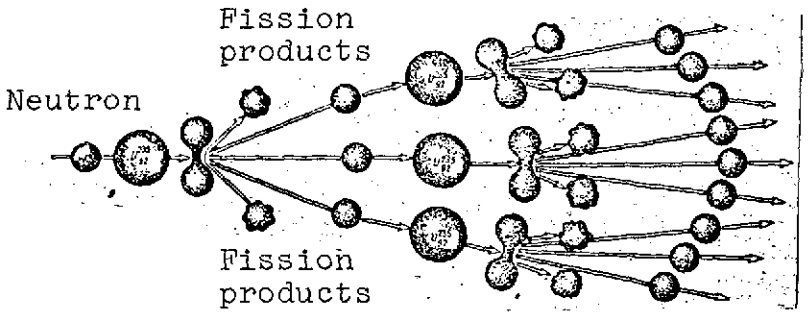

Figure 3. Chain reaction in uranium fission

In nuclear fission of uranium, approximately 83% of the energy is transformed into kinetic energy of the fragments; 3% is associated with the prompt fission gammas; 3% is carried off by neutrons at fission. The remaining 11% of the energy is carried off gradually by β-particles and gamma rays during radioactive decay of the isotopes (fragments) formed at fission. /22

The discovery in 1940 by the Soviet physicists G. N. Flerov and K. A. Petrzhak of the existence of a new form of radioactivity — spontaneous nuclear fission of the isotope U^{235} with a half-life T of approximately 10^{17} years — had important significance in the pursuit of the practical utilization of the chain reaction. Thus, extraneous neutrons are not required for the production of a fission chain reaction, since they are already available in uranium as a result of spontaneous fission.

Fission chain reactions may be achieved by fast, as well as by slow, neutrons in the bombardment of U^{235} isotope nuclei. Natural uranium is primarily a mixture of isotopes of U^{238} and U^{235}, with the composition of U^{235} being equal to only 0.7%. The U^{238} isotopes form the remainder. Therefore, in order to produce in practice a chain reaction, it is necessary to separate these isotopes, a task which, although possible, is a rather difficult one. This is due to the fact that U^{238} can be split only by neutrons with a higher energy than that of the neutrons produced in the fission of U^{235}. Thus,

neutrons formed in the fission of U^{235} with an energy on the order of 1 MeV are primarily scattered by U^{238}, which are present in much greater numbers. The energy of the neutrons decreases gradually until it reaches the energy corresponding to the resonance region (approximately 1 - 10 eV). In this energy region, the probability of neutron capture by U^{238} nuclei increases abruptly as opposed to capture by U^{235} nuclei. The fission chain reaction initiated in natural uranium is rapidly extinguished, since the neutrons are primarily captured by U^{238} nuclei without having had the opportunity to bring about fission of additional U^{235} nuclei. /23

U^{239} is formed in neutron capture by nuclei of U^{238}. In the process of β^--decay, the U^{239} is transformed into a new 93rd element Np^{239}. The half-life of U^{239} is equal to 23 minutes.

The isotope Np^{239} is also unstable. In the process of β^--decay (T = 2.3 days), it is transmuted into an element with atomic number 94, plutonium:

$$U_{92}^{238} + n_0^1 \rightarrow U_{92}^{239} + \gamma; \quad (11)$$

$$U_{92}^{239} \xrightarrow[T=23\ \text{min}]{\beta^-} Np_{93}^{239}; \quad (12)$$

$$Np_{93}^{239} \xrightarrow[T=2,3\ \text{day}]{\beta^-} Pu_{94}^{239}. \quad (13)$$

Plutonium is also radioactive. It is transformed by α-decay into the isotope U_{92}^{235}. The half-life of plutonium is 24,000 years.

Plutonium is of interest, since a chain reaction can be produced by the action of neutrons, just as in U^{235}. Thus, plutonium, along with U^{235}, is a nuclear fuel which is used to produce atomic energy.

Nuclei of uranium or plutonium, having captured a neutron, can divide in many ways (up to 30 - 40). The mass numbers of the fission products have values from 72 to 158. For example, isotopes of strontium, barium, lanthanum, cesium, iodine, zirconium, niobium, argon, xenon, and other elements are formed during fission. The most probable fission results in fragments with mass numbers of 95 and 139.

Most of the fission fragments are unstable, and as a result of one — at times of three — successive β-decays, are transformed into stable isotopes. This decay, for some fission products, is accompanied by gamma emission. The half-life of various fission products varies over a very wide range, from fractions of a second to many thousands of years.

Nuclear Reactor

In the brief survey of the development of nuclear physics from the first experiments of Rutherford to our times, we are not attempting to recreate the biography of the atom. Therefore, we will not touch upon a number of dramatic events, the raised hopes and the dashed expectations which followed from the moment of discovery of the first chain reaction to the control of the intranuclear energy. We will note only that the genius of man overcame the barriers placed by nature along the road to the secrets of the atom. At present, atomic energy is at the service of mankind. The controlled fission chain reaction of uranium or plutonium is carried out at special installations called nuclear reactors. /24

Before entering into a description of a nuclear reactor, we will briefly mention the conditions that must be created in order to achieve a self-sustaining chain reaction. The neutron multiplication factor in fissionable material must not be less than 1. The neutron multiplication factor gives the relative size of each neutron generation formed in the fission of U^{235} nuclei in comparison with the size of the preceding neutron generation.

In natural uranium, a chain reaction is impossible. However, if uranium is mixed with a material which effectively slows down fast neutrons formed in the fission of U^{235}, then the resonance capture of neutrons by U^{238} (i.e., in the energy range of 1 - 10 eV) will be reduced to a minimum. In this case, some of the neutrons slowed down to thermal energies, i.e., corresponding to the kinetic energies of molecular motion (approximately 0.025 eV), can produce the necessary number of U^{235} nuclei fissions to maintain the chain reaction. This occurs because the capture probability (fission cross section) by U^{235} nuclei increases for thermal neutrons. Neutrons which were not slowed to thermal energies will be captured by U^{238} nuclei.

The slowing down of nuclei occurs in collisions with nuclei of atoms of any substance. The most effective moderators of neutrons are the light elements whose nuclear mass differs insignificantly from that of neutrons (see Chapter 2). In addition, it is important that the neutron capture cross section of the moderator atoms be as small as possible. The best neutron moderators are heavy water, carbon (graphite), beryllium or beryllium oxide. Water has a relatively high thermal neutron capture cross section, and can be used as a moderator only in reactors operating with enriched uranium. /25

Thus, having brought together a specific amount of natural uranium and moderator, and having distributed them in a certain manner in space, we obtain an installation known as a nuclear reactor. Natural uranium and moderator are distributed in space in such a manner so that neutrons formed in the fission of U^{235} nuclei will not by wholly absorbed by nuclei of U^{238}, but that a portion will be slowed down to thermal energies ($\sim$ 0.025 eV), avoiding the resonance energy region (1 - 10 eV), and will split other U^{235} nuclei, giving rise to new neutrons. Under such conditions, a self-sustaining fission chain reaction will be developed. An accumulation of Pu^{239} occurs in the reactor according to the above-mentioned

reactions (11) - (13), since a portion of the neutrons are captured by U^{238} nuclei.

The part of the reactor which contains the nuclear fuel, and which is the site of a controlled chain fission reaction is known as the active zone of the reactor.

The control of the chain reaction in the reactor is achieved by special rods made from cadmium or boron. Both of these absorb neutrons very strongly and, by varying the position of the rods in the reactor, it is possible to accelerate or to slow down the nuclear reaction that is taking place there.

Depending upon the distribution of nuclear fuel and moderator in the active zone, the reactors may be divided into two large categories: homogeneous and heterogeneous. Homogeneous reactors are those in which the fuel and moderator form a homogeneous mixture in the form of a solution, alloy, chemical compound, or suspension. Reactors are heterogeneous if the nuclear fuel is distributed in the form of blocks surrounded by the moderator.

More than 30 years have passed from the initial operation of the first atomic reactor by Enrico Fermi, on the tennis court in the football stadium in Chicago. Since then, a great number of reactors of various types, differing in size as well as in capacity (from a fraction of a watt to millions of kilowatts) have been designed and built in many countries of the world. However, regardless of particular construction details, the basic plan of all types of reactors remains the same as that of the original atomic "kettle" (reactor), /26 as it was earlier called.

Depending upon their purpose, reactors may be subdivided into several types: experimental reactors designed for the investigation of new reactor construction methods and the development of various technological methods; breeder reactors used in the production of nuclear fuel, in particular, Pu^{239}; and power reactors used in the production of energy.

Intense neutron and gamma fluxes are formed in the operation of nuclear reactors. Therefore, in order to eliminate the adverse effect of radiation on the service personnel, reactors are surrounded by biological shields. Shielding materials include concrete, steel, polyethylene, water, or their combinations, depending upon the construction characteristics of the reactor. For example, Figure 4 gives the schematic cross section of the thermal, heavy water moderated power reactor of the Argonne National Laboratory, with a power generating capacity of 300 kW [4].

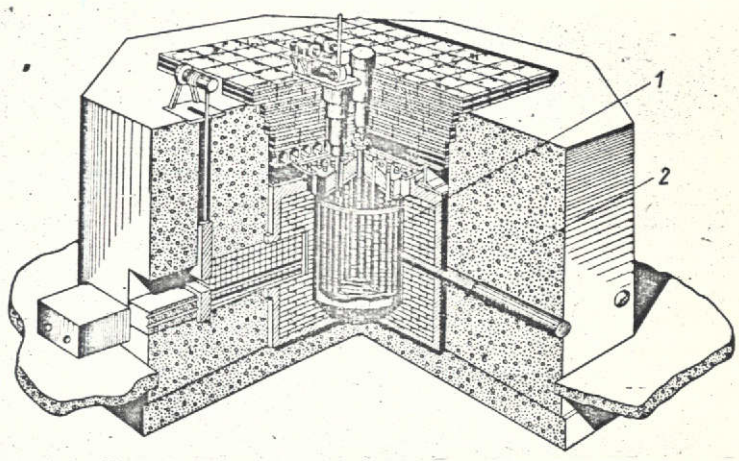

Figure 4. Argonne heavy water reactor: 1 — active zone; 2 — reactor cement shield with wall thicknesses of 2.6 m

/27

It is necessary to cool the nuclear reactor, since a large amount of energy is liberated during a chain reaction. If the reactor moderator is heavy water, it is continuously pumped out of the reactor into a heat exchanger, where it gives up its heat to plain water. Graphite reactors are cooled by the passage of water, other special liquids, or gases.

At the present time, the utilization of atomic energy, liberated by nuclear fission in reactors, is primarily by way of transformation of heat into electrical energy. To achieve this, the liquid or gas (called a coolant) at a high temperature at the reactor exit is passed through a heat exchanger (or steam generator). There, the coolant gives up the heat to water, transforming it into steam at a high temperature. The steam is directed at a turbine, connected to a generator of alternating current. Thus, the nuclear reactor replaces the boiler (Figure 5).

Undoubtedly, this method of practical utilization of nuclear energy is not the best. However, experience with atomic power plants and with nuclear powered vessels has demonstrated its unquestionable superiority over thermal engines.

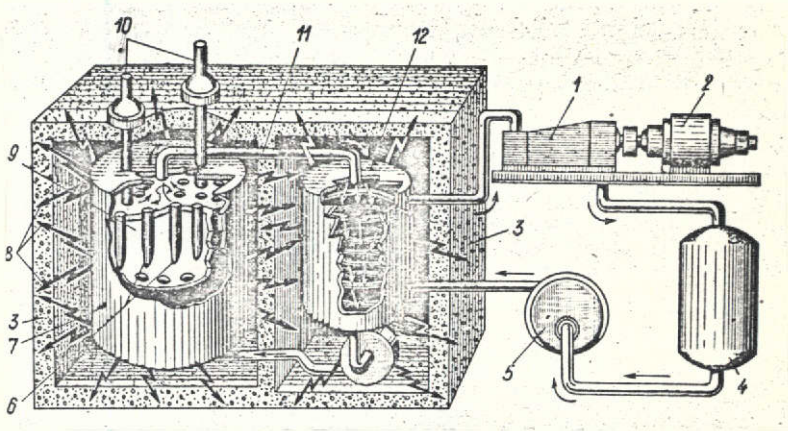

Figure 5. Schematic of an atomic electric power plant [5]:

1 — turbines; 2 — alternating current generator; 3 — concrete shield; 4 — condenser; 5 — circulation pump; 6 — uranium rods; 7 — reactor; 8 — gamma radiation produced in the active zone of the reactor; 9 — moderator; 10 — control rods; 11 — coolant; 12 — steam generator

/28

In addition, the safe utilization of similar installations has been demonstrated. At the present time, the growth of atomic power generating plants is increasing steadily. For instance, the capacity of atomic electric power plants has increased in the USSR by approximately 8 million kilowatts in the 9th fine-year plan.

The potentials of nuclear energy, as well as its economic feasibility, will increase even more when installations will be constructed capable of direct conversion of atomic energy into electrical energy.

In this regard, the investigation into the design of magnetohydrodynamic (MHD) generators is rather promising. Here, electrical energy is produced by the passage through magnetic fields of ionized gases, heated to high temperatures in a reactor. The efficiency of MHD generators can be 50 - 60%, while that of steam turbine electrical power stations reaches only 35 - 40% [1].

Nuclear reactors are not only sources of energy, but are also "producers" of isotopes. Thus, in the process of fission of the uranium nuclei, radioisotopes (products of the fission of uranium) are accumulated in the reactor. These are widely used in various

areas of science and technology. Moreover, stable elements placed in a reactor are transformed into artificial radioisotopes by the intense neutron fluxes.

Man is exposed to the action of radiation during all stages in the utilization of atomic energy, starting with the use of the nuclear reactor, the production of radioactive isotopes, and ending with the application of radioactive tracer methods.

In order to have a correct understanding of the effect of radiation on the organism, let us briefly examine the processes of interaction of radiation with matter.

CHAPTER 2

INTERACTION OF RADIATION WITH MATTER

Radiation produced during radioactive decay or during nuclear reactions passes through matter, interacting with atoms and molecules of the medium, imparting to them its energy.

The processes of interaction with matter take place in different /29
ways, depending upon the type of radiation. Therefore, it is worthwhile to consider separately the interaction with matter of charged particles (α- and β-particles), neutrons, and gamma photons.

Here, we will limit ourselves to a discussion of radiation with energies up to a few megaelectronvolts, and will not touch upon interaction with matter of very high energy radiation produced in modern accelerators.

Interaction of Charged Particles with Matter

Charged particles passing through matter gradually lose their energy as a result of interaction with atomic electrons, as well as with the electric field of the nucleus. In the process of interaction with atomic electrons, the kinetic energy of charged particles is dissipated in the ionization and excitation of the atoms of the medium (ionization losses). In the final analysis, the kinetic energy lost by the charged particles is transformed into thermal energy. Interacting with the electric field of the nucleus, the charged particle slows down and changes its direction of motion. As a result of such an interaction, bremsstrahlen radiation is produced. The decreases in the kinetic energies of the charged particles in the process of interaction with the electric field of the nucleus are radiative losses.

For electrons of low and intermediate energies (up to a few megaelectronvolts) and for α-particles and protons (for energies up to a few tens of megaelectronvolts), the primary energy losses are due to ionization and excitation of atoms of the medium. Two charged particles are formed in the ionization process: a positive ion (or atom) which has lost one or more electrons from the outer shell of the atom and a free electron. On the average, the energy expended in producing one ion pair in a gas does not depend on the energy of the charged particle over a wide range of energies.

The energy required to produce one ion pair in air is equal to, on the average, 35 eV for α-particles, and 34 eV for electrons. However, this quantity is not equal to the true ionization potential of the atom, which is 15 - 17 eV, i.e., the energy required to strip /30
an electron from an atom. The energy required to produce one ion pair is determined by the ratio of the energy of the primary particle and the average number of ion pairs formed by one particle over its entire path length. Thus, all types of energy losses (primarily due to ionization and excitation) are included in the indicated values, and are counted towards the formation of one ion pair. Approximately two or three excited atoms are produced for every ion pair formed.

Let us consider a parallel beam of α-particles. As the beam passes through matter, the energy of the α-particles is gradually dissipated by ionization losses. This process continues until the energy of the α-particle is reduced to such an extent that it is unable to produce ionization, at which time it will combine with two electrons to form a helium atom. The track of an α-particle is very straight, due to the fact that the mass of the α-particle is approximately 7000 times greater than that of the electron. If, in addition, we take into account that all α-particles emitted by a given radioactive isotope have, as a rule, one and the same energy, then the number of α-particles in a parallel beam passing through a unit area of an absorber will remain constant, and go abruptly to zero at the end of the range (Figure 6). The range is the minimum thickness of the absorber necessary for a complete absorption of the ionizing particles. The number of ion pairs formed by the particle over its entire path length in matter is known as the total ionization. Obviously, the total ionization:

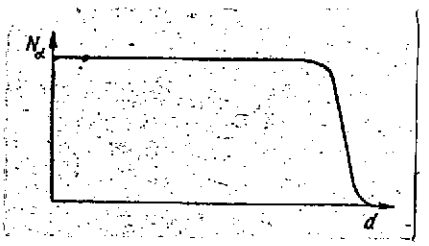

Figure 6. Typical absorption curve of an α-particle in matter:
N_α — number of α-particles passing through a 1 cm^2 area of matter per second; d — absorber thickness

$$I_{tot} = \frac{E}{w} \text{ ion pairs.} \qquad (14)$$

where E is the energy of the particle, w is the average energy required to produce one ion pair. For example, an α-particle with an energy of 3.5 MeV ($3.5 \cdot 10^6$ eV) produces a total ionization in air of:

$$I_{tot} = \frac{3.5 \cdot 10^6 \text{ eV}}{35 \text{ eV}} = 10^5 \text{ ion pairs,}$$

where 35 eV is the average energy spent by an α-particle to produce one ion pair. /31

The greater the energy of the particle, the greater is its range and, consequently, the greater the total ionization. Ranges of α-particles produced by radioactive materials attain 8 [-9] cm in air (Figure 7), and several tens of microns in soft biological tissues. Table 1 gives the ranges of α-particles in air and soft biological tissue for the most important radioactive isotopes.

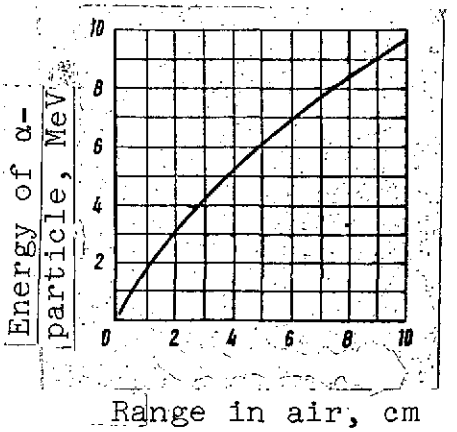

Figure 7. Dependence of the α-particle range in air on energy E_α

TABLE 1. APLHA PARTICLE RANGE IN VARIOUS MATERIALS

Radioactive isotope	Half-life	α-particle energy, MeV	Range in air, cm	Range in soft biological tissue, mm	Total ionization 10^5 ion pairs
Po^{210}	138.4 days	5.3	3.8	45	1.52
Po^{212}	$3 \cdot 10^{-7}$ sec	8.8	8.6	105	2.5
Ra^{226}	1622 years	4.8	3.3	40	1.37
Rn^{222}	3.83 days	5.5	4.0	49	1.57
Th^{232}	$1.4 \cdot 10^{10}$ yrs	4.0	2.5	31	1.14
U^{238}	$4.5 \cdot 10^{9}$ yrs	4.2	2.7	34	1.2
Pu^{239}	$2.4 \cdot 10^{4}$ yrs	5.15	3.7	43	1.47

It follows from Table 1 that the total ionization produced by α-particles is several hundred thousand ion pairs. Let us examine the variation in the specific ionization (linear ionization density), i.e., the number of ion pairs formed per unit path length as a function of α-particle penetration in matter.

It is apparent that the smaller the energy and, consequently, the particle velocity, the greater is the probability of its interaction with atomic electrons. Consequently the specific ionization /32

increases with increasing penetration in matter, reaching a maximum at the end of the range. Figure 8 gives the specific ionization as a function of α-particle energy.

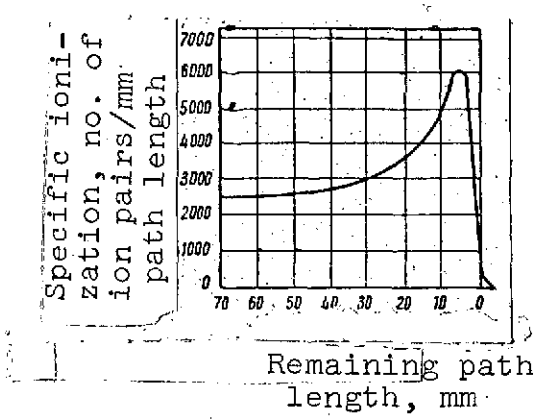

Figure 8. Variation in the specific ionization along the track of an α-particle

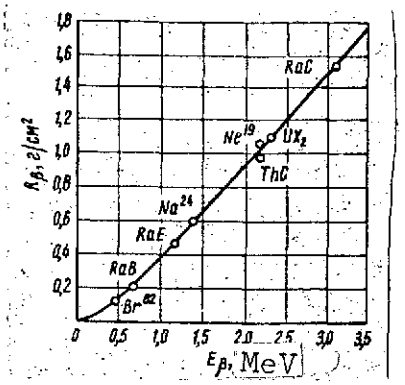

Figure 9. β-particle range, R_β, as a function of energy E_β

Let us consider α- and β-particles of identical energy. Since the mass of the β-particle is many times smaller than that of an α-particle, its velocity is greater and, as a result, the probability of its interaction with matter is significantly less than that of an α-particle. Furthermore, the charge on the β-particle is two times smaller than that on the α-particle, a fact which further determines a smaller probability of interaction. Consequently, the specific ionization of β-particles is smaller than that of α-particles. That is, a smaller number of ion pairs is formed per unit path length of a β-particle with a smaller fraction of energy loss. As a result, the range of a β-particle in matter is greater than that of an α-particle of equal energy. The specific ionization produced by an α-particle in air over a 1 cm path is, on the average, several tens of thousands of ion pairs, while for β-particles — it is only 50 - 100 ion pairs.

Figure 9 gives the dependence of the β-particle range R_β on its energy E_β. Table 2 gives the maximum ranges of β-particles in air, aluminum, and soft biological tissue.

TABLE 2. BETA PARTICLE RANGES IN VARIOUS MATERIALS

Radioactive isotope	Half-life	Maximum β-particle energy, MeV	Maximum range, cm		
			In air	In soft biological tissue	In aluminum
C^{14}	5570 yrs	0.155	22	0.02	0.008
Na^{24}	15.4 hrs	1.390	465	0.72	0.222
P^{32}	14.3 days	1.704	610	0.92	0.285
S^{35}	87.1 days	0.167	28	0.02	0.01
Ca^{45}	165 days	0.255	47	0.06	0.022
Co^{60}	5.26 yrs	0.310	62	0.09	0.029
Br^{82}	35.4 hrs	0.465	116	0.16	0.056
Sr^{89}	50.36 days	1.500	510	0.80	0.247
W^{185}	73.2 days	0.430	93	0.15	0.044
Tl^{204}	3.56 yrs	0.783	217	0.35	0.11

We can see that the range in air of β-particles with an energy of 1.5 MeV (Sr^{89}) is 510 cm, while the range of α-particles with an energy of 8.8 MeV does not exceed 9 cm (see Table 1). /33

Scattering has a greater effect on β-particles (electrons), since they are approximately 7000 times lighter than α-particles.

As a result of significant scattering, the path of β-particles in matter is not straight, and the true path of an electron in matter can be 1.5 - 4 times greater than its range. Decreasing the electron energy increases the scattering effect. As a result of successive scattering events, electrons may undergo significant angular deviations from the initial direction, and may even move in the opposite direction (backscatter). Therefore, even the absorption curve of monoenergetic electrons (i.e., electrons having one and the same energy) differs appreciably from alpha particle absorption curves. Electrons are scattered even in passing through the thinnest of

absorbers. As a result, in comparison with the initial beam, the intensity is decreased. Further along, the intensity of the beam decreases to zero as a result of scattering and ionization losses (Figure 10).

In view of the continuous nature of the β-spectrum, the absorption curve of a beam of β-particles decreases more rapidly than the absorption curve of monoenergetic electrons. This is due to the fact that β-particles of different energies are completely absorbed by different thicknesses of absorbers. The typical absorption curve of a parallel beam of β-particles is given in Figure 11. /34

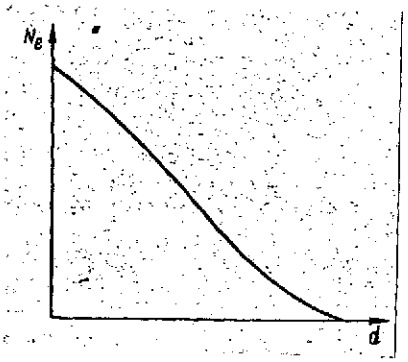

Figure 10 Typical absorption curve of monoenergetic electrons: N_e — number of electrons passing through a 1 cm^2 area of matter per second; d — absorber thickness

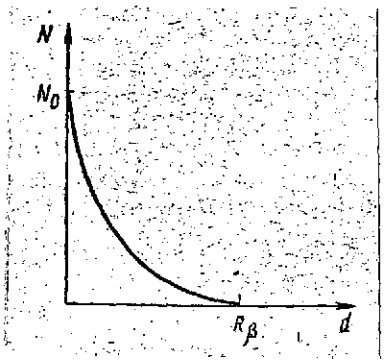

Figure 11. Typical absorption curve of β-particles in matter: R_β — maximum range; d — absorber thickness

Let N_0 be the number of β-particles passing through 1 cm^2 of absorber area per second. Then the number of particles transmitted through a thickness d cm of absorber is approximately equal to:

$$N = N_0 e^{-\mu d} \qquad (15)$$

μ characterizes the relative decrease in the flux of β-particles after transmission through an absorbing layer 1 cm thick.

The range of α- and β-particles in any given material is approximately as many times shorter or longer than their range in air as the density of the given material is correspondingly greater or

smaller than the density of air. For example, the range of α- or β-particles in soft biological tissues is approximately 770 times smaller than the range in air, since the density of these tissues ($\rho \approx 1$ gm/cm^3) is approximately 770 times greater than the density of air (0.001293 g/cm^3).

Very frequently, the thickness of the absorber is expressed not in centimeters or millimeters, but in units of gm/cm^2 or mg/cm^2, i.e, indicating the mass of the absorber corresponding to 1 cm^2 of its surface.

If the thickness of the absorber is known in these units, then /35
in order to determine its thickness in centimeters, it is necessary to divide the indicated value by the density of the given absorber. Thus, the range of the β-particles is 0.54 gm/cm^2. Let us determine its range in centimeters in aluminum, whose density is equal to 2.7 gm/cm^3:

$$\frac{0.54 \text{ g/cm}^2}{2.7 \text{ g/cm}^2} = 0.2 \text{ cm}$$

The convenience of measuring absorber thicknesses in the above-mentioned units consists in the fact that ranges of α- or β-particles of a given energy have approximately one and the same values (in units of gm/cm^2) in different materials.

Interaction of Neutrons with Matter

In passing through matter, neutrons representing a flux of uncharged particles interact only with atomic nuclei. This process is analogous to the collision of two billiard balls. A billiard ball moving at high speed will transfer part of its energy to a stationary ball on colliding with it, and will change its original direction of motion. From the laws of mechanics, it is known that the greater the mass of the stationary ball, the smaller the fraction of energy that

will be transferred to it during the collision. If the masses of the colliding balls are equal, then the average loss by the moving ball per collision will be half of its energy.

Neutrons, having a definite energy reserve, interact with nuclei of matter in an analogous manner, transferring part of their energy and changing their direction of motion. This process is known as elastic scattering. Atomic nuclei which receive as a result of the collision a specific amount of kinetic energy (recoil nuclei) are "ejected" from the electron shell and pass through matter producing ionization (since they are charged). The smaller the mass of the nuclei of the medium which is traversed by the neutrons, the greater the fraction of energy that is lost by neutrons in the process of elastic scattering. Neutrons lose, on the average, half of their energy per scattering collision with hydrogen nuclei. With carbon nuclei, this is reduced to approximately 14 - 17%, and with argon nuclei — to not more than 8 - 9%. Therefore, as mentioned previously, /36 light materials such as deuterium, plain or heavy water, carbon, paraffin, or beryllium make the best neutron moderators.

In the process of elastic scattering, the energy of the neutron gradually decreases and approaches the energy of thermal motion of the atoms and molecules of the medium, equal to approximately 0.025 eV. These are thermal neutrons. The number of collisions with hydrogen nuclei must be equal to 2^n in order to slow neutrons to thermal energies from an initial energy of 1 MeV. That is:

$$\frac{10^6\ \text{eV}}{0{,}025\ \text{eV}} = 2^n,$$

It follows that $4 \cdot 10^7 = 2^n$, and n = 25, i.e., 25 collisions will be required. The neutron energy is reduced to 0.025 eV after 100 collisions in carbon, while 2100 collisions are required in interactions with uranium nuclei. This process is completed within 10^{-6} seconds. The trajectory of neutrons in matter is given by a broken line with straight portions 8 - 10 cm in length. The straight portion is ~ 1 cm if the energy is reduced to 100 eV.

Thermal neutrons will continue to wander in matter until their capture by one of the atomic nuclei. As a result, the following nuclear reaction takes place:

$$X_Z^A + n_0^1 \rightarrow X_Z^{A+1} + \gamma. \qquad (16)$$

A product isotope is formed with the excess energy of the nucleus given up in the form of a gamma photon as a result of this reorganization. This type of interaction is called radiative capture. Artificial radioactive isotopes are obtained by means of the above-mentioned reaction in nuclear reactors where intense neutronfluxes are produced.

Prior to capture by the nucleus, the thermal neutron experiences approximately 50 collisions in the medium. The lifetime of a thermal neutron is $2 \cdot 10^{-4}$ seconds. Not only thermal but also fast neutrons may be captured by atomic nuclei. As a result, a nuclear reaction will occur, and an alpha particle, proton, or neutron may be ejected from the nucleus. The excess energy is given up in the form of a gamma photon: /37

$$\left.\begin{aligned} X_Z^A + n_0^1 &\rightarrow X_{Z-1}^A + p_1^1 + \gamma; \\ X_Z^A + n_0^1 &\rightarrow X_{Z-2}^{A-3} + \alpha + \gamma; \\ X_Z^A + n_0^1 &\rightarrow X_Z^A + n_0^1 + \gamma. \end{aligned}\right\} \qquad (17)$$

A nucleus of a new element is formed by means of the first two reactions, while as a result of the third reaction, we will again obtain the initial nucleus, but with an emitted neutron of different energy. Therefore, a reaction of the third type is called inelastic scattering. The process of inelastic scattering has a relatively large probability for many atomic nuclei from the middle or end of the periodic table of elements. The probability of neutron capture by nuclei with the formation of the isotope of the initial element is inversely proportional to its velocity, or its $E^{1/2}$ (E is the neutron energy). Comparing the process with the collision of billiard balls, it is apparent that the smaller the kinetic energy (velocity) of the moving ball, the greater the probability that it will come to a stop, after collision, in the vicinity of the stationary ball, as if

sticking to it. However, if the neutron energy is close to one of the energy levels of the excited nucleus, then the probability of neutron capture by the nucleus increases sharply. This process is called resonance capture. One of the reactions of the above-mentioned type takes place as a result of this capture.

The probability of various types of neutron interactions with nuclei depends on their energy. Therefore, neutrons have been divided arbitrarily into three groups: fast neutrons (500 keV < E < 10 MeV); intermediate neutrons (0.5 eV < E < 500 keV), and thermal neutrons ($E \leq 0.025$ eV).

The primary interaction process for fast neutrons is elastic scattering, although other interaction processes are also possible with a significantly reduced probability (inelastic scattering, nuclear reactions, and radiative capture).

With increasing energy, the probability of inelastic, in comparison with elastic, scattering increases from 10 MeV until the role of inelastic scattering becomes greater than that of elastic scattering in neutron interactions with matter for $E \geq 20$ MeV.

Inelastic scattering as well as radiative capture are the most characteristic modes of interaction for intermediate neutrons. It should be noted that the probability of resonance capture sharply increases in the energy range of intermediate neutrons. Radiative capture is the most probable interaction for thermal neutrons. /38

The concept of "effective cross section" σ, measured in units of cm^2, is introduced to characterize the probability of interaction of neutrons with nuclei. The effective cross section of elastic scattering is equal to $1 - 3 \cdot 10^{-24}$ cm^2, and increases with increasing atomic number of the medium. Let us examine the interpretation of the concept of "effective cross section". Let us assume that the elastic scattering $\sigma = 3 \cdot 10^{-24}$ cm^2. This implies that if $3 \cdot 10^{23}$

nuclei are distributed over an area of 1 cm^2, then each neutron which passes through this surface experiences an elastic scattering interaction. Or, if only one nucleus is located per 1 cm^2, then $3 \cdot 10^{23}$ neutrons have to pass through this surface in order to produce a single elastic scattering process. The effective cross section for thermal neutrons varies over a wide range for different elements, from $\sim 10^{-24}$ to 10^{-21} cm^2.

Let us consider a narrow beam of neutrons falling perpendicularly on the surface of a layer of matter. Let N_0 be the number of neutrons incident on 1 cm^2 per second. After passage through a certain thickness of matter, the number of neutrons will decrease as a result of various interaction processes. The attenuation law of a neutron beam by a layer of matter with thickness d cm can be given by:

$$N_d = N_0 e^{-n\sigma d}, \tag{18}$$

where N_d — neutron beam after transmission through a layer d thick; n — number of nuclei per cm^3 of matter; σ — effective neutron interaction cross section with nuclei. The effective interaction cross section is the sum of all possible types of interactions (inelastic and elastic scattering, radiative capture, etc.). The average distance traversed by fast neutrons in matter between two collisions is known as the mean free path of scattering λ_s. The thick- of matter which reduces the neutron flux as a result of absorption by a factor of e (e = 2.718) is known as the mean absorption length /39
λ_a. It is assumed in this case that scattering does not occur. Sometimes this quantity is known as the relaxation length.

Summing up, it can be noted that either charged (recoil) particles* which produce ionization directly or gamma photons which produce ionization by means of secondary particles are formed in all interaction processes of neutrons with matter.

*Charged particles (protons and alpha particles) are also formed during nuclear reactions.

Interaction of Gamma Rays with Matter

In the process of transmission through matter, gamma photons interact with atomic electrons, nuclear electric fields, as well as with protons and neutrons constituting the nucleus. These interactions produce the attenuation of the intensity of the original beam of gamma rays. The beam intensity is defined as the energy flux per second, per 1 cm^2 of area perpendicular to the direction of propagation of the beam. The intensity is measured in units of ergs/(cm^2 · sec), MeV/(cm^2 · sec) or, correspondingly, by keV/(cm^2 · sec), eV/(cm^2 · sec).

If the beam of radiation is made up of photons of one and the same energy, then the radiation is monoenergetic. In this case, the intensity is given by:

$$I = N h\nu, \tag{19}$$

where N is the number of photons passing through a 1 cm^2 area per second. If the radiation beam is made up of photons of different, but completely discrete, energies $h\nu_1$, $h\nu_2$, $h\nu_3$, etc., then the spectrum of such radiation is called discrete. In other words, a discrete spectrum is made up of a number of monoenergetic spectral lines. The intensity of a discrete spectrum is given by:

$$I = N_1 h\nu_1 + N_2 h\nu_2 + N_3 h\nu_3 + \ldots + N_n h\nu_n. \tag{20}$$

where N_1, N_2, H_3, ..., N_n is the number of photons passing through a 1 cm^2 area per second, having energies of $h\nu_1$, $h\nu_2$, $h\nu_3$, ..., $h\nu_n$.

Nuclear protons and neutrons, just as atomic electrons, may have well defined energy states. As a result, the spectrum of γ-rays, just as that of characteristic rays, is discrete. /40

In the interaction of γ-rays with atomic electrons, the decreased intensity is due primarily to the photoelectric absorption, incoherent (Compton) scattering, and pair production.

Nuclear reactions of various types are produced as a result of interaction with nuclear protons and nuutrons. These are called photonuclear reactions.

Photoelectric absorption. Photons can transfer all of their energy to atomic electrons in the process of interaction. As a result, the photon disappears, and its energy is expended in stripping the atom of an electron, which also receives kinetic energy. In this process, the energy of the photon, as a rule, is transferred to the electrons closest to the nucleus, i.e., to those in the K-shell. The probability of photoelectric absorption depends strongly on the atomic number Z of the material. With increased atomic number of the absorber, the probability of photoelectric absorption increases proportionately to Z^4. The probability of photoelectric absorption sharply decreases with increasing energy.

Incoherent (Compton) scattering. This interaction process is analogous to the collision of two billiard balls. That is, the photons interacting with atomic electrons transfer to them part of their energy, while being scattered at some given angle. In contrast with photoelectric absorption, when the photon interacts primarily with K-shell electrons, in incoherent scattering, photons interact with outer (valence) electrons whose binding energy is minimal.

The reduced beam intensity, as a result of incoherent scattering, is due to the fact that photons interacting with electrons of the medium are scattered in various direction, and leave the initial beam. In addition, photons transfer part of their energy to the electrons.

With increasing energy of the radiation, the probability of incoherent scattering decreases, but more slowly than that of photoelectric absorption. The probability of the process increases with

increasing atomic number of the absorber, proportionally to Z, i.e., /41
approximately proportionally to the absorber density.

Pair production. In the process of photon interaction with electric fields of atomic nuclei of the absorber, two particles — an electron and a positron — are formed. This is accompanied by the disappearance, or, as it is called, by the annihilation of the photon.

Pair production occurs only if the photon energy exceeds the sum of the equivalent energy of the rest mass of the electron and positron. It is well known that the rest masses of the electron and positron are the same, and are equal to 0.511 MeV. Thus, pair production takes place only when the photon energy exceeds 1.022 MeV.

The probability of pair production increases with increasing photon energies, and is proportional to Z^2 of the absorber.

Thus, in all interaction processes between γ-photons and matter, part of the energy of the radiation is transformed into kinetic energy of electrons which pass through matter, producing ionization. The probability of each of the indicated processes of interaction of γ-rays with matter depends on the energy of the γ-photons and atomic number of the absorber.

Photoelectric absorption is important for low energy γ-photons and in materials with high atomic number. At γ-ray energies on the order of 60 keV in air, water, and biological tissues, photoelectric absorption constitutes 50% of the total absorption. At an energy of 120 keV, photoelectric absorption constitutes only 10%. At energies above 200 keV, this process may be neglected since, in this case, the primary process of interaction which reduces the intensity of γ-rays in matter is incoherent scattering. For materials of intermediate atomic number (iron, copper), the portion due to photoelectric absorption becomes insignificant for energies greater than 0.5

MeV. For lead, it becomes necessary to take into account photoelectric absorption up to energies on the order of 1.5 - 2.5 MeV.

Pair production begins to have some significance in matter with small Z at approximately 10 MeV, while for matter with high Z (lead) — at 3.5 MeV, and higher.

In view of the fact that the energy of γ-photons emitted by most of the radioactive isotopes does not exceed 2 MeV, the most important processes of interaction for the solution of practical problems of protection are photoelectric absorption and incoherent scattering. The attenuation of γ-rays in matter is greater for photons of lower energies and for absorber of higher density and atomic number. If I_0 denotes the intensity of the radiation at a point K in the absence of an absorber (Figure 12a), then the intensity of the γ-radiation beam after traversing a layer of the medium d cm thick (Figure 12b) will be equal to:

/42

$$I = I_0 e^{-\mu d}, \qquad (21)$$

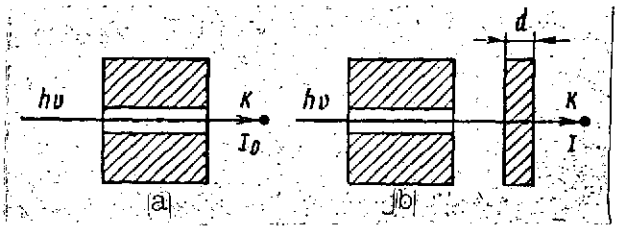

Figure 12. Diagram of a narrow beam of γ radiation passing through matter.

where μ is the linear attenuation coefficient (cm^{-1}) characterizing the relative attenuation of γ-radiation in a layer of the medium 1 cm thick.

Figure 13 gives the attenuation curve of γ-radiation beam as a function of absorber thickness. It is clear that the curve does not intersect the abscissa. Instead, it monotonically approaches the axis for large thicknesses. This implies that whatever the thickness of a material, it is impossible to completely absorb the γ-ray flux. It is only possible to reduce its intensity by a given factor. This represents a material difference in the nature of attenuation between γ-photons and α- and β-particles, since for the latter it is always possible to select a thickness of matter which will completely absorb the flux of α- or β-particles.

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It follows from Figure 13 that the intensity of the beam is decreased by a factor of two, i.e., is 0.5 I_0 after traversing a layer of matter $\Delta_{0.5}$ thick. The thickness of the absorber necessary to reduce the flux of the γ-radiation by one half is known as the gamma-ray half-value layer. The following relationship exists between the linear attenuation coefficient and the half-value layer:

$$\Delta_{0.5} = \frac{0{,}693}{\mu}. \qquad (22)$$

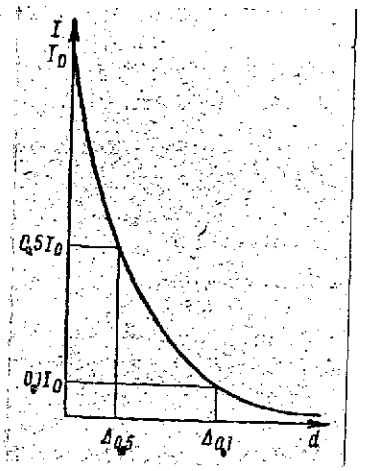

Figure 13. Dependence of attenuation of γ-radiation intensity I on absorber thickness

Knowing the half-value layer, it is rather easy to determine the thickness of absorber necessary to attenuate radiation by a given factor. For example, one half-value layer reduces the intensity of radiation by two, two layers by four, three layers by eight, four layers by 16, n layers by 2^n. Thus, for example, in order to reduce the intensity by a factor of 128, it is necessary to take a number of n half-value layers, such that:

$$2^n = 128.$$

In other words, it is necessary to determine the number of times 2 is a multiplier in 128. It is easy to show that in this case, n = 7, i.e., seven half-value layers will reduce the intensity of radiation by a factor of 128.

The thickness of the absorber necessary to reduce the intensity of radiation by a factor of 10, i.e., I = 0.1 I_0, is known as the tenth-value layer, and is denoted by $\Delta_{0.1}$ (see Figure 13). It is clear that n tenth-value layers of the absorber are required to reduce the radiation by a factor of 10^n. The following relationship holds between the tenth-value layer and the linear attenuation coefficient:

$$\Delta_{0,1} = \frac{2,3}{\mu}. \qquad (23)$$

The above-mentioned formula (21) gives the attenuation law for a narrow γ-beam. In this case, experimental conditions are such that the radiation detector located behind the absorber does not record scattered photons. Narrow beam geometry may be produced only under laboratory conditions by means of a specially collimated beam of radiation. In practice, there are usually no special collimators between the radiation detector and absorber. Scattered photons impinge on the recording detector (Figure 14). In this case, a broad beam geometry is produced. /44

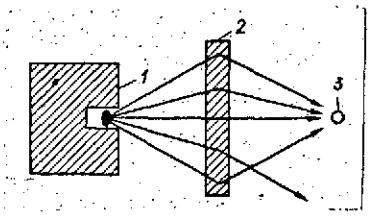

Figure 14. Diagram of γ-radiation broad-beam geometry:
1 — radiation source; 2 — absorber; 3 — detector

It is clear that for a given thickness of absorber, a smaller attenuation of the radiation intensity will occur under conditions of broad beam geometry. The attenuation for broad beam geometry is given by:

$$I = I_0 e^{-\mu d} B(h\nu, \mu d, Z), \qquad (24)$$

where B(hν, μd, Z) is the buildup factor which takes into account the change in the intensity of γ-photons due to scatter. The buildup factor depends on the energy of radiation, thickness, and atomic number of the absorber.

Under broad-beam geometry, photons traversing matter experience one or more scattering interactions, and fall on the detector instrumentation at various angles. It is clear that in this case, it is rather difficult to speak about the intensity of radiation. It follows from the definition that the intensity is the energy passing per second through a 1 cm^2 area perpendicular to the direction of propagation of the radiation. Consequently, in order to determine the intensity of scattered radiation at some point in space, it is necessary to know the angular distribution of scattered photons. This distribution depends not only on the energy of radiation and

thickness of the absorber, but also on the configuration of the absorber and detector of radiation, and their relative location in space. Therefore, in the investigation of attenuation laws of a broad beam of radiation, it is more convenient to use the concept of dose (see Chapter 3) and not intensity, especially since most of the dosimetry instrumentation measures dose rather than intensity. /45
If P_0 is the dose rate (dose per unit time) of radiation at a given point in the absence of an absorber, while P is the dose rate of radiation at the same point after passage through a thickness d of the absorber, then:

$$P = P_0 e^{-\mu d} B_D(h\nu, \mu d, Z) \quad (25)$$

where $B_D(h\nu, \mu d, Z)$ is the dose buildup factor, which takes into account the contribution of the scattered radiation to the dose rate after passage of the γ-radiation through the absorber under broad beam geometry.

Figure 15 and Table 3 give examples of the dose buildup factor /46
in water and lead for γ-photons of various energies [6 - 8].

To a sufficient degree of accuracy, the dose buildup factor can be represented by:

$$B_D(h\nu, \mu d, Z) = A_1 e^{-\alpha_1 \mu d} + A_2 e^{-\alpha_2 \mu d}, \quad (26)$$

where A_1, $A_2 = 1 - A_1$, α_1, α_2 are constants dependent upon the energy of radiation and atomic number of the absorber [6, 9, 10].

Substituting the analytical expression for the buildup factor in Formula (25), an equation will be obtained for the magnitude of the dose of γ-radiation, after passage through a thickness d of absorber, which also takes into account scattered radiation:

$$P = P_0 \left[A_1 e^{-\mu (1+\alpha_1) d} + A_2 e^{-\mu (1+\alpha_2) d}\right]. \quad (27)$$

As mentioned previously, the attenuation of the radiation intensity is due primarily to photoelectric absorption, incoherent scattering, and pair production. Therefore, the linear attenuation coefficient μ can be represented by:

$$\mu = \tau + \sigma + \chi, \qquad (28)$$

where τ, σ, χ are the relative decreases in radiation intensity due to photoelectric absorption, incoherent scattering, and pair production, respectively.

In the region of radiation where pair production does not play a significant role, for example, for E_γ up to 3 MeV for lead, and up to 10 - 20 MeV for water, aluminum, and biological tissues, we have:

$$\mu = \sigma + \tau. \qquad (29)$$

For light elements (water, aluminum, biological tissues), photoelectric absorption may be neglected above 200 keV:

$$\mu = \sigma. \qquad (30)$$

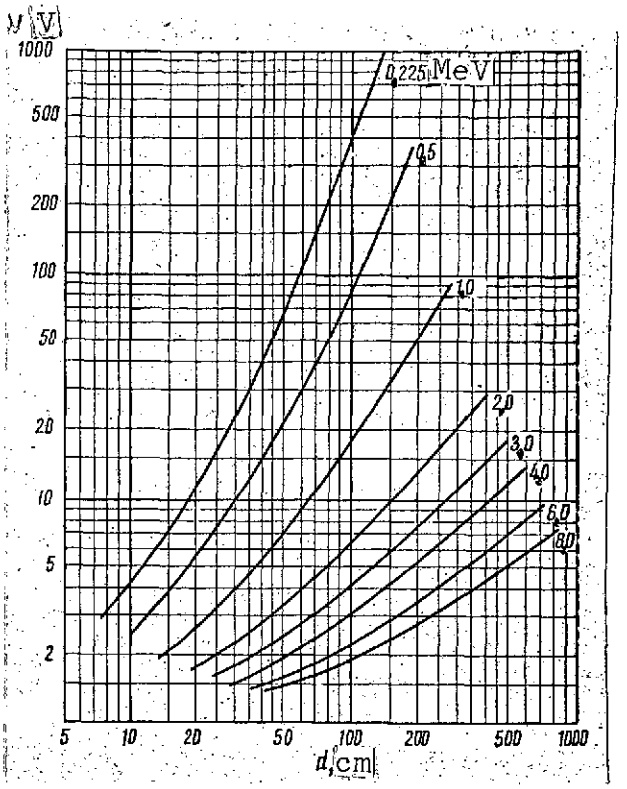

Figure 15. Dependence of dose buildup factor B on a layer of water, d thick, for a broad beam of γ-radiation of various energies

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TABLE 3. DOSE BUILDUP FACTORS IN LEAD FOR VARIOUS μd (μ — LINEAR ATTENUATION COEFFICIENT cm^{-1}, d ABSORBER THICKNESS, cm)*

Energy of radiation, hν, MeV	μd				
	2	4	7	10	15
0,5	1,42	1,69	2,00	2,27	2,65
1,0	1,69	2,26	3,02	3,74	4,81
2,0	1,76	2,51	3,66	4,84	6,86
3,0	1,68	2,43	3,75	5,30	8,44
4,0	1,56	2,25	3,61	5,44	9,80
5,11	1,46	2,08	3,44	5,55	11,74
6,0	1,40	1,97	3,34	5,69	13,80
8,0	1,30	1,74	2,89	5,07	14,05
10,0	1,23	1,58	2,52	4,34	12,54

*Translator's note. Commas in numbers represent decimal points.

i.e., attenuation of radiation is due to incoherent scattering only.

Figure 16 gives curves showing the variation of μ, σ, χ, and τ for lead, and their dependence on energy of γ-radiation. With increasing energy of γ-radiation, the attenuation coefficient μ initially decreases, passes through a flat minimum at an energy approximately equal to 2.5 - 3.5 MeV, and then gradually increases in the energy region where pair production begins to play a major role. Since χ is proportional to approximately the square of the atomic number of the absorber, the increase of the attenuation coefficient μ of light elements (with low Z) begins at significantly higher energies. /48

For water and aluminum, some increase in μ begins at energies on the order of 30 - 40 MeV.

For monoenergetic radiation, the attenuation coefficient depends only on the energy of radiation, atomic number, and density of the absorber, and does not depend on its thickness.

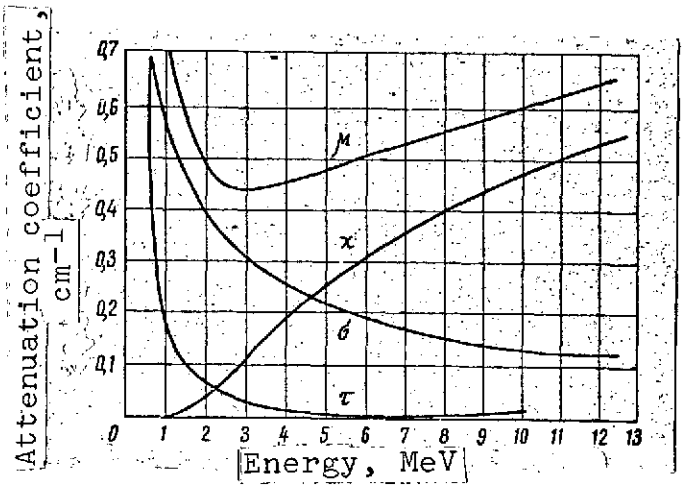

Figure 16. Dependence of μ, τ, σ, and χ of lead on energy of γ-radiation:

If, however, the radiation spectrum is not monoenergetic, i.e., it is constituted by photons of various energies, then this radiation may be considered monoenergetic with some effective energy E_{eff} and effective attenuation coefficient μ_{eff}. Attenuation laws are still given by the same equations valid for monoenergetic radiation. However, in contrast with μ, the effective attenuation coefficient μ_{eff} depends not only on the energy of radiation and type of absorber, but also on its thickness. This is due to the fact that various components of the non-monoenergetic spectrum of radiation are absorbed

differently in passing through the absorber. Consequently, the spectral composition of radiation changes with thickness of the absorber. Since photons of low energies are attenuated more strongly, the portion of higher energy photons increases with increasing thickness. The hardness of the radiation increases while μ_{eff} correspondingly decreases. Figure 17 shows the nature of the change in the effective linear attenuation coefficient μ_{eff} and γ-radiation from the radioactive isotopes Ir^{192} and radium as a function of thickness of a lead absorber [6].

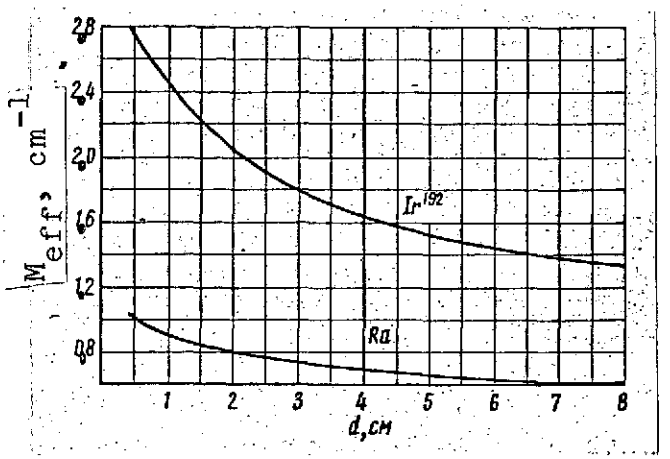

Figure 17. Variation of the effective attenuation coefficient μ_{eff} as a function of lead absorber thickness d for γ-radiation from Ir^{192} and a radium source in equilibrium with its daughter decay products

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The values of attenuation coefficients, as well as those of coefficients A, α_1, and α_2 in the analytical expression for buildup factors of γ-radiation of various energies and various absorbers are given in many books and reference handbooks [6 - 11].

CHAPTER 3

RADIATION DOSE AND BIOLOGICAL EFFECT

The living organism is a product of the environment, being continually subjected to the action of favorable, as well as unfavorable, external factors. This circumstance contributed by evolution to the formation in living organisms of various adaptive mechanisms which allowed them to become adapted to external influences. It also led to the formation of receptors which reacted in a definite manner to the actions of the surrounding medium, and helped mobilize the defensive capabilities of the organism.

The adaptive mechanisms formed by evolution in living organisms, including man, provide for a normal life activity and interaction with the external environment.

However, situations may arise in nature or during the course of productive activity of man when the action of one or more unfavorable factors may sharply increase. The defensive system of the organism may be unable to cope with these factors, leading to a disruption of its normal life activity or to its destruction.

Uncovering the secrets of nature, man learned to determine and to differentiate a series of unfavorable factors by their nature and degree of action on various systems of the organism. Man developed and continues to develop complex defensive and prophylactic measures which either eliminate or weaken these factors.

In the development of defensive measures, it is important to know that normal level of action which will not have adverse effects on a given individual or population. As a result, efforts of scientists are directed not only toward the study of the effects of various adverse factors on living organisms, but also on the scientific substantiation of the permissible threshold levels of action /50 which will not produce adverse effects.

Soon after the discovery of x-rays and radioactivity, it was determined that radiation is not without effect on man, and that under certain conditions, it brings about serious irreversible processes in the organism, having at times lethal results. Naturally, these facts attracted the attention of scientists to the investigation of the mechanism of action of this new harmful agent, to the establishment of the principles of action of ionizing radiation on living organisms, and to the determination of permissible levels of radiation.

Biological Action of Ionizing Radiation and Relative Biological Effectiveness

Penetrating through matter, α- and β-particles, γ-photons, neutrons, protons, recoil nuclei, and heavily charged ions, usually known as penetrating or ionizing radiation, expend their energy primarily by ionization and excitation. It has been established that approximately two or three atoms are excited for each ion pair produced. It should be noted that charged particles (α- and β-particles, protons, recoil nuclei, heavily charged ions) produce ionization directly, while neutrons and γ-photons do not produce ionization directly. The latter, in the process of interaction with matter, produce charged particles (recoil nuclei or electrons) which receive all or part of their energy during each particular interaction. The charged particles formed during the interaction produce ionization in passing through matter.

It is known that the lifetime of ionized and excited atoms is negligibly short — on the order of 10^{-6} sec. Positive and negative ions, colliding together, recombine (rejoin) forming neutral atoms or molecules. In simple substances, whose molecules are made up of atoms of one and the same element, molecules of the original substances are formed during recombination, and no changes are produced in their chemical constitution as a result of ionization. /51

However, the ionization or excitation of complex molecules may lead either to their destruction or to the formation of chemically active free radicals with unsaturated bonds. The latter may initiate a number of chemical reactions as a result of which new chemical compounds will be formed. For example, in aqueous solutions, atomic hydrogen and OH, HO_2, and other radicals are formed. Thus, in complex substances, changes in their chemical properties occur as a result of the action of radiation, either directly (disintegration of molecules) or indirectly (formation of free radicals). The more complex the matter, the more complex are the processes produced in it.

In biological substances, the formation of free radicals or the disintegration of molecules leads to changes in biochemical processes in organisms. As a result, exchange processes are disrupted, enzyme activity is suppressed, and tissue growth is either slowed or stopped. Thus, the vital activity of isolated systems or of the organism as a whole may be disrupted by the action of ionizing radiation. It is clear that the greater the number of ionizing events occurring in an organism as a result of radiation, the greater will be its biological effect. Consequently, the biological action of radiation depends on the number of ion pairs formed or on the directly related quantity — the absorption of energy.

However, it is not only the total energy absorption of radiation that determines the biological effect of the ionizing radiation. Numerous experiments have shown that the biological effect increases with the linear energy transfer (LET) of the radiation. That is, it depends on the linear density of ionization, i.e., on the number of ion pairs formed per unit path length of the radiation in matter. For example, if the eye is irradiated by a flux of electrons and neutrons which produce the same ionization in the crystalline lens, cataracts will be formed in the lens irradiated by neutrons, since /52 the latter produce recoil protons having a greater specific ionization than electrons. As a result, the concept of relative biological effectiveness (RBE) was introduced to evaluate the biological effect of various types of radiation.

The greatest amount of data on the biological effect of various types of radiation was initially obtained from the irradiation of animals by x-rays with an energy of up to 250 keV, and from occupation exposure of radiologists. It was decided, therefore, to estimate the biological effect of radiation by comparison with the biological effect produced by x-rays and gamma rays of the indicated energies. In this case, the specific ionization produced by secondary electrons is ∿ 100 ion pairs per 1 μm path length in water. The corresponding LET is 3.5 keV per 1 μm in water [12].

It was later shown that the biological effectiveness of x- or γ-radiation is practically independent of the energy of the photon. Therefore, the RBE of x- or γ-radiation of all energies is equal to 1. It is clear that the RBE of β-particles (electrons) and γ-photons is the same, since the ionizing effect of γ-photons is produced by secondary electrons formed in the process of interaction of γ-radiation with matter.

Thus, RBE is a number which shows how much greater or smaller the expected biological effect of a given type of radiation is than the biological effect of x- or γ-radiation under similar conditions of irradiation and equal energy absorption in the irradiated substance.

It is considered that the RBE of fast neutrons is equal to that of α-particles and protons. This is due to the fact that hydrogen atoms constitute more than 90% of all the atoms in biological tissues. As a result, the action of neutrons on an organism is due to ionization produced primarily by recoil hydrogen nuclei, i.e., protons.

It was mentioned in Chapter 2 that thermal neutrons are captured by atomic nuclei in their passage through matter. As a result, either stable or radioactive isotopes are formed. In biological tissue, the two most important reactions which determine the action of thermal neutrons are: /53

$$H_1^1 + n_0^1 \rightarrow H_1^2 + \gamma; \quad (31)$$

$$N_7^{14} + n_0^1 \rightarrow C_3^{14} + H_1^1. \quad (32)$$

As a result of Reaction (31), hydrogen nuclei capture thermal neutrons and are transformed into nuclei of heavy water, deuterium, with the emission of γ-photons. Reaction (32) leads to the formation of carbon nuclei with the emission of a hydrogen nucleus — a proton of a definite energy. C_6^{14} nuclei are radioactive, emitting β-particles, and turn into a stable isotope of nitrogen:

$$C_3^{14} \rightarrow e_{-1} + N_7^{14}. \quad (33)$$

Thus, the ionization of tissue by the action of thermal neutrons is due to γ-photons and protons (hydrogen nuclei). Since the cross sections for these reactions are different, the amount of nitrogen in tissue is much less than that of hydrogen. The specific ionization is not the same either due to the γ-photon and proton. Thus, the RBE turns out to be equal to 3 for thermal neutrons.

It is clear that the RBE depends not only on the type of radiation, but also on its energy. For example, α-particles or protons of a few MeV in energy produce a significantly greater specific ionization in matter than at an energy of a few hundred MeV.

It should be noted that the RBE depends also on the duration of irradiation. Experimental data show that the RBE is greater for long-term irradiation, rather than for short-term. In localized irradiation, the RBE depends also on which organ or groups of organs are exposed to radiation. We do not have the opportunity to discuss here in detail the various aspects related with the problem of the RBE and to analyze numerous experimental data accumulated up to the present time. This is not the objective of this book. It is clear that additional, more solid justification for the maximum permissible levels of radiation and the correct estimate of the effects of radiation depend substantially on the validity of the selected RBE value for various types of radiation and their energies.

Since the RBE depends not only on the LET, but also on several biological parameters which are difficult to estimate, it was decided not to use this concept in the comparison of various types of energies according to the type of expected biological effect depending only upon the specific ionization. It was decided to leave the concept to the radiologists [12]. /54

According to the recommendations of the International Commission on Radiation Protection, the concept of a quality factor QF was introduced to estimate the effectiveness of radiation producing various specific ionizations in tissues. This factor indicates how much greater is the expected biological effect than that produced by radiation with an LET equal to 3.5 keV per μm path length in water. The relation between the LET and the quality factor is given in Table 4 [11 - 13].

TABLE 4. DEPENDENCE OF THE QUALITY FACTOR QF ON THE SPECIFIC IONIZATION AND ON LET*

Average Specific ionization, ion pairs/μm water	LET in water, keV/μm	Quality factor QF
⩽100	3,5	1
100—200	3—7,0	1—2
200—650	7,0—23	2—5
650—1500	23—53	5—10
1500—5000	53—175	10—20

For heavy recoil nuclei, the LET can exceed 175 keV/μm. However, experimental data shows that even in this case the quality factor QF will not exceed 20. Therefore, the quality factor for heavy nuclei is also considered to be 20.

Under continuous, whole body irradiation at small dose levels (see Chapter 4) within permissible values, when the biological effect practically does not depend on the dose rate of irradiation, the

value of the RBE does not exceed the quality factor, and may be made equal to it. Table 5 gives the values of QF and RBE for various types of radiation for continuous irradiation [12, 13].

TABLE 5. QUALITY FACTOR QF AND RBE UNDER CONTINUOUS WHOLE BODY IRRADIATION (WITH THE EXCEPTION OF THE LENS OF THE EYE)

Type of irradiation	QF and RBE	Type of irradiation	QF and RBE
γ-radiation	1	Thermal neutrons	3.0
X-rays	1	" " , 5 keV	2.5
Electrons	1	" " , 20 keV	5.0
Positrons	1	" " , 100 keV	8.0
β-particles	10	" " , 500 keV	10.0
α-particles (E ≤ 10 MeV)	10	" " , 1 MeV	10.5
Protons (E ≤ 10 MeV)	10	" " , 5 MeV	7.5
Heavy recoil nuclei	20	" " , 10 MeV	6.5

Since the crystalline lens is especially sensitive to certain radiation (neutrons, α-particles) with high LET, it is recommended that a special value of QF be used, higher than that given in Table 5. In particular, for fast neutrons with energy 0.5 - 5 MeV, QF is equal to 30 instead of the usual 10, as it follows from Table 5. Quality factors used to evaluate possible consequences of crystalline lens irradiation are given in Table 6 [12, 13]. /55

TABLE 6. QUALITY FACTOR QF FOR NEUTRONS IN THE IRRADIATION OF THE LENS OF THE EYE BY NEUTRONS OF VARIOUS ENERGIES

E, keV	Thermal	5	20	100	500	1000	5000	10,000
Quality factor QF	10	8	15	24	30	30	20	20

It should be noted that the critical organ for the eye, in the external irradiation by fluxes of weakly penetrating radiations (α- and β-particles, electrons), is the crystalline lens located at a depth of 300 mg/cm^2. The critical organ for the skin is the basal layer, which is located at a depth of 7 mg/cm^2. The thickness of the skin and subcutaneous tissues is 100 mg/cm^2.

Absorbed Dose

Changes produced in the irradiated object under the action of various types of radiation depend on the absorbed energy. Therefore, the most convenient characteristic of radiation which determines the degree of its effect on an organism is the absorbed energy of the radiation. /56

At the VII International Congress of Radiologists (Copenhagen, 1953), it was recommended that the energy of any type of radiation absorbed per 1 gram of matter be called absorbed dose, and that the unit of absorbed dose be the rad.

The rad is the unit of absorbed dose and is equal to 100 ergs of absorbed energy per 1 gram of any substance, independent of the type and energy of the ionizing radiation. Multiples of the unit are the millirad (mrad) and microrad (μrad)

$$1 \text{ mrad} = 10^{-3} \text{ rad} = 0.1 \text{ erg/g},$$
$$1 \text{ μrad} = 10^{-6} \text{ rad} = 0.0001 \text{ erg/g}.$$

The absorbed dose per unit time is known as the absorbed dose rate, and is given in units of rad/hr, rad/min, rad/sec, or in other multiples.

The absorbed dose may be measured by the effects produced by radiation in the process of interaction with matter. In order to do this, it is necessary to establish a direct relationship between the

changes in the physical or chemical properties of the radiation detector (ionization, changes in conductivity, intensity of luminescence, change in color, etc.), and the absorption of energy of the radiation.

It is rather easy to measure the absorbed dose for charged particles from the ionization or from other radiation effects produced in matter. This is due to the fact that the energy of the radiation is absorbed at the location at which ionization occurs. Moreover, if the particle flux and the energy falling on the irradiated volume is known, then the absorbed energy in the entire volume will be given by the product of the number of particles and their energy, as long as the particle range is not greater than the irradiated volume. Accordingly, the mean absorbed does is equal to:

$$D = \frac{NEt \cdot 1.6 \cdot 10^{-6}}{100\, d\rho} \text{ rad}, \qquad (34)$$

where N is the number of particles falling on a 1 cm^2 surface of the irradiated volume per second; E is the energy of the particles in MeV (for β-particles with a continuous spectrum, the mean energy is understood); t is the time of irradiation in seconds; $1.6 \cdot 10^{-6}$ is the energy equivalent of the unit, MeV (1 MeV = $1.6 \cdot 10^{-6}$ ergs); 100 is the energy equivalent of the rad (1 rad = 100 ergs/g); d is the range of the ionizing particle in the irradiated material in cm; ρ is the density of the material in g/cm^3. /57

We consider here a mean absorbed dose, since the specific ionization along the particle track is not constant.

By an analogous method, it is possible to determine the absorbed dose produced by fast neutrons, since the ranges of the recoil protons are very short and the absorption of the energy of the neutrons and the ionization produced by the recoil protons occurs practically at one and the same spot. It is more difficult to measure the absorbed dose due to x- and γ-radiation. Let us consider a volume A, next to the surface of the irradiated object (Figure 18). A γ-photon,

which is absorbed in the volume A, transfers its energy to the electron, whose range may be such that it travels outside this volume. Consequently, ionization produced by this electron in the volume A will not be directly related to the absorption of the energy of the γ-photon. At the same time, other electrons formed as a result of absorption of photons outside the volume A — for example, in air or in adjoining regions of the irradiated body — may produce ionization in this volume. Thus, we cannot form an opinion about the magnitude of the absorbed γ-radiation energy in Volume A based on the ionization measured in the same volume.

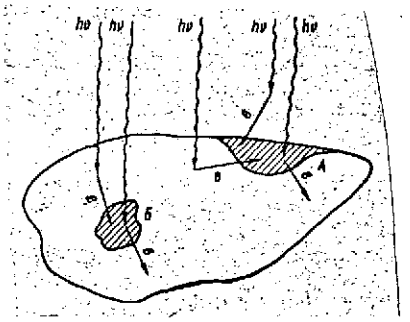

Figure 18. Diagram for determining the absorbed dose from γ-radiation

Let us now examine the volume B (see Figure 18) which lies inside the irradiated body at a distance from its surface equal to or greater than the range of the secondary electrons. Moreover, not a single electron formed outside the irradiated object falls into the volume B. Ionization produced along the track of secondary electrons outside the volume B, where they were produced as a result of the absorption of γ-photons, will be exactly compensated by the ionization produced by the secondary electrons generated in the irradiated object surrounding volume B. In this case, from the radiation measured in volume B, we can reach a conclusion about the magnitude of the absorption of γ-radiation energy in the given volume. Consequently, the absorbed dose of γ-radiation can be measured from the ionization in the air cavity inside the irradiated object if the cavity is surrounded by a layer of matter with thickness equal to or greater than the maximum range of secondary electrons, i.e., as long as electronic equilibrium holds. /58

Exposure Dose

Of all types of radiation, x- and γ-rays were the first to find practical applications, specifically in biology and in medicine. While ionization was historically the earliest and the best-documented

method of detecting radiation, it is still widely used today. We attempted to determine what unit of radiation measurement would make it possible to connect the ionozation effect with absorbed radiation energy. In 1928 the Roentgen (R) [14] was adopted as this unit.

We shall not discuss the definitions and determinations of the Roentgen used previously. Let us just mention that the Roentgen was considered as a quantity of radiation characterizing the absorption of radiation energy per unit mass of air. This interpretation contradicted the definition, and introduced difficulties in the evaluation of absorbed energy. These difficulties were bypassed by means of various reservations and corrections. The inconvenience of the Roentgen as a unit used to estimate the absorption of radiation energy was, in addition, compounded by the fact that this unit was introduced only for x- and γ-radiation. In order to carry out a comparative study of effects produced by the action of other forms of radiation (α- and β-particles, neutrons, etc.), it became necessary to introduce the concept of "roentgen equivalent physical"*. This created further inconveniences, and made necessary a reconsideration of the existing terminology, leading to the introduction of the rad, a new unit of absorbed dose, universal for all types of radiation and directly related to effects produced by radiation. /59

As a result of the fact that ionization methods of measurement are still widely used in practice, and since nearly all dosimetric instrumentation is calibrated in Roentgens, the unit was not eliminated from the arsenal of metrological units. As a result, the Roentgen was retained as a unit, along with the rad, in GOST.

In contrast with the accepted definition, the Roentgen is considered not as a unit of absorbed energy, but only as a unit of determining the ionizing capacity of x- and γ-radiation in air.

The quantization of x- and γ-radiation, based on their ionizing action in terms of the total electric charge of ions of the same

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*Roentgen equivalent physical (rep) is the dose of any type of ionizing radiation for which the absorption of energy per 1 gram of a medium is equal to the energy lost in the ionization produced in 1 gram of air by a dose of 1 R of x- or γ-radiation. For air, 1 rep is equal to 87 ergs/gm, while for muscle — it is 93 ergs/gm.

sign produced per unit volume of air, is known as the exposure dose of x- and γ-radiation.

The Roentgen (R) is the unit of the exposure does of x- or γ-radiation which produces 0.001293 grams of air by means of the associated secondary particles ion pairs equivalent to 1 electrostatic unit (1 cgse) of each sign (0.001293 g is the mass of 1 cm^3 atmospheric air at 0° C at a pressure of 760 mm Hg).

Submultiple units are the milliroentgen (mR) and microroentgen (μR):

$$1\ R = 10^3\ mR = 10^6\ \mu R.$$

The exposure dose rate is the dose per unit time. It is measured in units of R/hr, R/sec, R/sec, etc.

Since the charge on the electron is equal to $4.8 \cdot 10^{-10}$ esu, the number of ion pairs, n_i, formed by an exposure of 1 R in 0.001293 /60 grams of air is equal to:

$$n_i = \frac{1\ \text{cgse}}{4.8 \cdot 10^{-10}\ \text{cgse}} = 2.08 \cdot 10^9. \qquad (35)$$

It is known that, on the average, 34 eV are required to produce a single ion pair in air. Consequently, at an expsoure of 1 R, the energy expended in ionization by secondary electrons in 0.001293 grams of air is equal to:

$$E = 2{,}08 \cdot 10^9 \cdot 34 = 7{,}07 \cdot 10^{10}\ \text{eV} = 0.113\ \text{erg},$$

or in 1 gram of air:

$$E = \frac{0{,}113}{0{,}001293} = 87\ \text{erg/g}.$$

A direct relationship between the exposure dose measured in Roentgens and the absorbed dose may be established only if the

radiation exposure dose is measured in an air volume surrounded by a layer of air, or air equivalent material, with a thickness greater than or equal to the range of secondary electrons (i.e., when the condition of electronic equilibrium holds). In this case, for an exposure of 1 R, the absorbed energy in air is equal to 87 erg/g, that is, the absorbed dose is equal to 0.87 rads.

It makes sense to talk about electronic equilibrium only when the range of secondary electrons is significantly smaller than the mean free path of the photon*. If these values become comparable, it is impossible to reproduce a Roentgen. As a result, the use of the Roentgen as a unit of exposure is limited to the measurement of x- or γ-radiation of up to 3 MeV in energy.

The absorption of energy in water and muscle tissue differs (by 4 - 10%) from the absorption of energy in air, due to the fact that Z_{eff} of water and muscle tissue is not exactly equal to Z_{eff} of air. In the energy interval from 150 keV to 3 MeV, the absorbed energy in water and muscle tissue is equal to 93 ergs/g, if the exposure in air, measured under conditions of electronic equilibrium, is equal to 1 R.

For bone, with a Z greater than that of air and consequently w with a substantially greater photoelectric absorption at low energies, the absorption of energy changes from 474 to 88 ergs/g, with increasing energy from 10 to 200 keV. Starting from 200 keV, the absorbed energy remains approximately constant and equal to about 88 ergs/g for an exposure of 1 R. Thus, having measured the ionization in air, under conditions of electronic equilibrium, we can form an opinion about the absorption of energy in biological tissues.

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*The mean free path of a photon is the thickness of the absorber that will reduce the intensity of γ-radiation by a factor of e.

Equivalent Dose

The concept of equivalent dose D_E is introduced in order to estimate the radiation hazard due to prolonged irradiation by different types of radiation. Its unit is the rem, which is the absorbed dose of any type of radiation which, under prolonged irradiation, will bring about the same biological effect as that produced by 1 rad of x- or gamma-radiation. The doses delivered by various types of radiation of equal rem will, under similar conditions of irradiation, be equivalent in biological effect. It is clear that between the absorbed dose D, expressed in rads, and the equivalent dose DE, expressed in rem, the following relationship exists:

$$D_E = QF \cdot D, \qquad (36)$$

where QF is the quality factor of the radiation. As previously indicated, this quantity defines the dependence of the biological effect of prolonged irradiation of the organism by a given type of ionizing radiation on the LET of the radiation. For β-particles and γ-photons (QF = 1), 1 rem is equivalent in effect to 1 rad; while for α-particles and protons (QF = 10), 1 rem is equivalent in biological effect to 10 rad of x- or γ-radiation, etc.

For a comparative estimate of the effects produced by the entry of radioactive materials into an organism, with the irradiation of internal organs, in addition to the quality factor QF, a distribution factor DF is introduced for osteotropic isotopes. This factor takes into account the influence of inhomogeneous distribution of radioactive isotopes and their carcinogenic effectiveness in relation to /62
Ra^{226}. In this case:

$$D_E = D \cdot QF \cdot DF \ldots . \qquad (37)$$

The dotted line indicates that other modifying factors may also occur, which take into account the characteristics of the distribution of isotopes in the organism.

It is clear that, under chronic exposure to ionizing radiation of mixed composition, the equivalent dose DE, expressed in units of rem characterizing an expected biological effect, is given by:

$$D_E = \Sigma D_i \cdot QF_i \cdot DF_i, \qquad (38)$$

where D_i is the absorbed dose of a given type of radiation in rads; QF_i and DF_i are the quality and distribution factors corresponding to the same type of radiation.

Relation Between Exposure Produced by γ-Radiation From Radioactive Substances and Their Activity

The number, as well as the energy, of emitted γ-photons varies over wide limits for various radioactive isotopes. Therefore, it is expeditious to know the relationship between the activity of a source and the exposure from γ-radiation from a given radioactive isotope.

For a point source* with an activity C, mCi, the exposure D, after a time t hours at a distance of R cm, is given by the formula:

$$D = \frac{\Gamma \cdot Ct}{R^2} \rho, \qquad (39)$$

where Γ is the Γ-constant of the isotope which determines the dose rate in units of R/hr produced by γ-rays of a given radioactive isotope at a distance of 1 cm from a point source whose activity is equal to 1 mCi.

It has been experimentally determined that the Γ-constant is 8.4 R/hr for radium in radioactive equilibrium with its short-lived decay products and enclosed in a platinum filter 0.5 mm thick. This implies that the point source of radium in a platinum filter 0.5 mm thick, with an activity of 1 mCi, in equilibrium with its decay /63

*A source is a point source if the dose or intensity of radiation at a given point is inversely proportional to the square of the distance to the source. The source may be considered to be a point source if its linear dimensions are 5 - 10 times smaller than the distance to the point of measurement of the dose or intensity.

products, produces an exposure rate equal to 8.4 R/hr at a distance of 1 cm from the source.

The Γ-constant for Co^{60} is 13.5 R/hr, for Cs^{137} — 4.2 R/hr; Ir^{192} —5.46 R/hr [7].

A comparison of Γ-constants of radium and CO^{60} shows that 1 mCi of Co^{60} produces exposures 1.6 times greater than 1 mCi of radium which is in equilibrium with its decay products and is enclosed by a platinum filter with a wall 0.5 mm thick. In other words, the radiation exposure in air produced by 1 mCi of Co^{60} is equivalent to approximately 1.6 mCi of radium, that is, γ-photons emitted by a Co^{60} preparation with an activity of 0.625 mCi produce the same radiation exposure as γ-radiation from 1 mCi of radium.

Very frequently, a so-called γ-equivalent of a radioactive equivalent is used to characterize γ-radiation from isotopes. This is measured in milligram-equivalents of radium. A γ-equivalent of a radioactive substance is equal to 1 mg-equiv of Ra if the γ-radiation from the substance under identical measurement conditions produces the same ionization as the γ-radiation from 1 mg (1 mCi) of radium in equilibrium with its short-lived decay products and enclosed in a platinum filter 0.5 mm thick. For example, if the γ-equivalent of this preparation is equal to 30 mg-equiv of Ra, this means that γ-radiation from this preparation will produce the same radiation exposures under specific measurement conditions as 30 mg radium in equilibrium with its decay products and enclosed in a platinum filter 0.5 mm thick.

When the Γ-constant of an isotope is known, the following simple formulas may be used to pass from activities measured in millicuries to γ-equivalents measured in milligram-equivalents, and vice-versa:

$$M = C\frac{\Gamma}{8{,}4}; \qquad (40)$$

$$C = M\frac{8.4}{\Gamma}. \qquad (41)$$

Here, M is the γ-equivalent of a given preparation in mg-equiv Ra (or gm-equiv Ra); C is the activity in mCi or Ci; Γ is its Γ-constant, while 8.4 is the Γ-constant for radium. /64

If the γ-equivalent of a substance is given in units of mg-equiv Ra, then Formula (3) becomes:

$$D = \frac{8.4\,Mt}{R^2}\mathrm{R}. \qquad (42)$$

Some Data Regarding Biological Consequences of Irradiation

As indicated earlier, the disruption of the vital activity of various organs and tissues takes place under the action of ionizing radiation on an organism.

Radiation-induced changes occurring in an organism may appear clinically either a relative short time after irradiation (hours, days) — acute radiation effects, or after an extended time period (years or even tens of years) — the so-called late effects. Moreover, the disruption of the structural elements responsible for heredity may occur in the organism as a result of radiation. In the majority of cases, these changes, while being harmless to the adult individual, may be harmful to future generations.

Therefore, in the estimation of radiation hazards to which various contingents of people or entire populations may be exposed, radiation effects are usually differentiated into somatic and genetic effects.

Somatic effects are those changes in the health of a given individual which occur as a result of irradiation. Somatic effects are manifested by acute or chronic radiation sickness, local radiation damage of various organs or tissues, and also by reactions of the organism to irradiation.

Acute radiation effects characteristically show a relation between radiation dose and reaction of the organism. Moreover, acute radiation effects show a threshold, i.e., they appear after some radiation dose is exceeded. /65

The great amount of material collected to date from experimental animal data, as well as from the correlation of the numerous data regarding the health status of roentgenologists, radiologists, and other individuals who were exposed to radiation, shows that no changes can be discovered in the health status of an individual exposed to a single whole body dose of up to 25 rem. No changes are observed in the blood, which is the first to show the effects of radiation.

External signs of radiation exposure are also absent for whole body doses between 25 - 50 rem. However, temporary changes, which quickly return to normal, may be observed in the blood.

Exposure to a dose of 50 - 100 rem leads to a feeling of fatigue, without a serious loss of capacity for work. Fewer than 10% of the irradiated may show signs of nausea, and moderate changes in the composition of the blood are observed. The condition of health soon returns to normal.

Single exposures to doses greater than 100 rem lead to various forms of acute radiation sickness.

Thus, exposure to a dose of 150 - 200 rem produces a short term moderate form of acute radiation sickness. It shows up as a long-lasting lymphopenia. Nausea may be observed in 30 - 50% of the cases in the first 24 hours after irradiation. Lethal outcomes do no occur.

Radiation sickness of a moderate severity arises from exposures to doses of 250 - 400 rem. Nausea and vomiting are observed in almost all irradiated individuals within the first day after irradiation. The number of leucocytes is sharply decreased, and subcutaneous hemorrhages occur. A lethal outcome may follow in 20% of the cases. Death follows 2 - 6 weeks after irradiation.

An acute form of radiation sickness develops after irradiation at a dose of 400 - 700 rem. Death may follow in 50% of the cases, one month after irradiation.

The most severe type of acute radiation sickness is observed after irradiation to doses greater than 700 rem. Two to four hours after irradiation, nausea appears. Leucocytes disappear almost immediately from the blood, subcutaneous hemorrhages take place and bloody diarrhea occurs. The mortality is 100%. Infectious diseases /66 and blood losses are the most frequent causes of death [15, 16].

The data presented about the consequences of the action of radiation apply to cases without subsequent therapeutic intervention. A number of radiation protective agents, which permit a significant reduction in the effects of radiation, are presently available, and experience dealing with the complex therapy of radiation sickness has been accumulated.

The clinical picture of radiation sickness of various degrees of severity as a function of radiation dose refers to single whole body exposures. If the irradiation to the same dose is carried out not once, but is extended in time, then the effect of irradiation will be decreased. This is related with the fact that living organisms — among them, man — are capable of reestablishing normal life activity after disruptions of one kind or another. Currently available experimental data allow us to assume that the recovery rate from radiation damage amounts to only 2.5% of the accumulated dose per day. Furthermore, the irreversible part of the damage is approximately 10% of the received dose [17]. For example, if an individual were exposed to a dose of 200 rem, then after 40 days the residual changes in the organism will be equal to those produced by an irradiation of 20 rem. If the individual is again irradiated to a dose of 200 rem, the damage will be equal to that produced by a dose of 200 + 20 = 220 rem.

The concept of effective dose, which takes into account the recovery process, is introduced to estimate the effect of prolonged

irradiation. Naturally, it is less than the total received dose for the same period.

Chronic radiation sickness may develop during systematic, repeated exposures to dose levels not causing radiation sickness, but at levels significantly higher than the maximum permissible levels (see Chapter 4). The most characteristic signs of chronic radiation sickness are the changes in the composition of blood (decrease in the number of leucocytes and anemia), and a number of symptoms of nervous system irregularities.

Symptoms of chronic radiation sickness are non-specific, and sometimes occur in illnesses due to other causes [15, 16].

According to well-established biological data, the reaction of /67
the organism to radiation may be late (after 10 - 20 years). These reactions may include leucosis, malignant neoplasms in various organs and tissues, cataracts, skin damage, life-span shortening (aging, leading to premature death unrelated to any specific cause). As a result of the non-specificity of the late effects of radiation, it is rather difficult and even impossible to correlate them with previous irradiation.

It is perfectly clear that for a specific person irradiated to a dose which would not bring about acute radiation damage, it is practically impossible to establish a causal relationship between irradiation and the appearance, for example, of leukemia, since this illness may be caused by other adverse, non-radiation connected factors. Even if, as a result of irradiation, some changes in the health status of this individual are reversed by treatment, it is still difficult to ascertain directly that the malignant neoplasm formed in this individual at a later date is specifically radiation-induced and is not caused by other environmental factors. Such a category, just as in the case of life expectancy, is probabilistic in nature and may be applied only to large population groups, and not to any given individual.

The scale of late effects may be evaluated by examining only the probability of occurrence of somatic effects in large population groups. Organism radiation effects which are evaluated by means of statistical methods are known as somatic-stochastic.

Experimentally obtained animal data, as well as long term observation of Hiroshima and Nagasaki atomic bomb victims or radiation therapy patients, show that late effects are due to rather large doses of radiation. For instance, the frequency of the occurrence of lung cancer among Hiroshima and Nagasaki victims increased only at radiation doses greater than 70 rem, while the frequency of occurrence of leukemia among adults exposed to doses greater than 100 rem is 0.2 - 0.5% [18].

In the solution of problems of radiation protection and of the levels of utilization of radioactive substances in therapy or in diagnostics, and in the examination of other aspects of the application of atomic energy in the national economy, it is extremely important to know the probability of occurrence of late effects for irradiation at low doses (a few rads). However, the evaluation of somatic-stochastic effects at low levels of radiation presents serious difficulties. Statistical treatment of data dealing with the health of a group of people irradiated for prolonged periods (roentgenologists and radiologists, and occupational workers) does not indicate any reliable differences in the duration of life or in the probability of malignant neoplasm occurrence in this group of individuals in comparison with the part of the population not exposed to radiation [12, 18, 19]. It is impossible to uncover changes in the health status, as well as any reliable somatic-stochastic effects, in individuals undergoing x-ray procedures, even though the associated radiation doses are tens and hundreds of time greater than natural background. For example, in the x-ray examination of the stomach, the radiation dose is 1.5 - 3 rem, of the lungs — 0.04 - 0.25 rem, shoulder — 0.7 - 1.0 rem [20]. It may be stated that at existing levels of knowledge of the biological effects of radiation, it is not possible to discover its harmful effects, including late effects, on the health of people exposed for long periods to doses

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exceeding 10 - 100 times the natural background (i.e., 1 - 10 rem per year). This refers to the irradiation of adults, without consideration of possible genetic consequences [12].

Taking into account the inadequate knowledge regarding the action of low radiation doses, the Scientific Committee of the UN and the International Commission on Radiation Protection (ICRP) recommend the extrapolation of data dealing with late effects due to large doses, to the low dose area by means of a linear dependence between radiation dose and biological effect [18, 21].

The assumption of a linear dependence of "dose-effect" for low radiation levels leads to the conclusion that there is no radiation threshold, i.e., the absence of harmless radiation doses, including those produced by natural background. Consequently, the extrapolated estimates of the expected late effects, based on the recommended concept, result in a reevaluation of the probable risk due to irradiation at low doses. Furthermore, this extrapolation does not take /69 into account the dependence of the radiation effect on the rate of irradiation and the influence of the recovery processes on the development of late effects.

Analysis carried out in this manner (let us note again that the risks are overestimated) show that if 10^6 individuals are exposed to a dose of 1 rem, then one may expect the appearance of 1 - 2 cases of leukemia per year [18, 19]. It should be noted that the natural frequency of all types of leukemias are, on the average, 50 cases per year per 10^6 persons [19, 22, 23]. In order to produce an equal number of leukemias as a result of irradiation, the dose delivered to 10^6 persons must be not less than 50 rem.

The natural frequency of tumors of various organs (excluding leukemia) is 2000 cases per year per 10^6 persons [22]. At the same time, extrapolated estimates show that in the irradiation of 10^6

persons to a dose of 1 rem, only 2 - 4 cases of malignant neoplasms will be observed [18, 19].

Thus, the frequency of occurrence of harmful late effects (leukemia, malignant neoplasms of various organs) as a result of low dose (on the order of a few rem) irradiation of the population is significantly smaller than the natural frequency of these same diseases associated with non-radiation causes.

As mentioned previously, genetic structures may be damaged by radiation. As a result of this, adverse radiation effects may appear in succeeding generations. Genetic effects will be observed only if the defective gene is crossed with another one having the same defect. Therefore, the probability of genetic effects produced by radiation depends not only on the irradiation dose, but on the number of persons in the entire population exposed to radiation.

The greater the total population and the smaller the number of individuals belonging to this population who are exposed to radiation, the smaller is the probability of matings between individuals who were exposed to radiation. Consequently, the probability of radiation-induced effects is also smaller.

The analysis of available data, as well as estimates made by various international and national agencies, shows that a dose greater than 6 - 10 rem per person in the population is genetically significant [12]. /70 This implies that if the gonadal dose from the moment of conception to age 30 (average human reproductive age) per person of the population is smaller than 6 - 10 rem from all sources of radiation produced by man (including radiation for medical purposes, and excluding natural background), then the additional damage as a result of genetic effects may be considered to be permissible and justifiable, based on the advantages which may be expected from increased practical application of atomic energy [12, 21].

The concept of a group dose, expressed in units of person · rem, was introduced to measure the effect of low dose radiation on large population groups. For example, if 10^6 persons are exposed to a dose of 3 rem, on the average, the group dose is given by $3 \cdot 10^6$ person · rem.

The concept of risk factor, expressed in the number of cases of leukemia, of malignant neoplasms in other organs, per population dose of 10^6 person · rem, is introduced to estimate the adverse late effects which may arise as a result of irradiation of significant population groups. The calculation of these factors, as previously mentioned, is made on the assumption that there is no dose threshold below which biological effects of radiation will not be observed [18].

Somatic, stochastic, and genetic estimates of the risk due to radiation are especially important in planning for the development of atomic energy, in the distribution of atomic materials, and in the development of the requirements and standards for providing radiation safety of atomic electric plants and other nuclear installations. Indeed, a preliminary estimate of possible levels of radiation, as well as the number of persons who may be exposed to radiation under normal use or due to emergency situations at one or another nuclear installation (for example, nuclear power plants) makes it possible to estimate the number of unfavorable results. As a result, specific requirements may be set on the location of the installation itself in relation to populated areas, waste disposal, protective shields, reliability of various elements, and so on.

CHAPTER 4

MAXIMUM PERMISSIBLE LEVELS OF RADIATION

Action of Ionizing Radiation on the Human Organism

In working with radioactive materials or sources of ionizing radiation (x-ray machines, accelerators, nuclear reactors) man may be exposed to radioactive emissions. Furthermore, it is necessary to differentiate between external irradiation and the possibility of the entry of radioactive materials inside the organism. /71

External radiation implies the action of radiation on a person when the source of radiation is outside the organism and there is no possibility of the radioactive substance getting into the organism. This takes place, for example, in working with x-ray machines and accelerators and in work with radioactive substances in hermetically sealed capsules (gamma installations for nondestructive testing of metals and radiation therapy of malignant neoplasms or powerful isotope sources for radiation sterilization in the irradiation of seeds prior to planting, etc.).

In certain cases, contamination of surfaces and air of work buildings, as well as contamination of hands and special apparel of of workers, is possible by radioactive materials while conducting radiochemical or other operations with radioactive materials (specifically in the separation of isotopes from a mixture of fission products, the preparation of radioactive sources,or compounds, the packaging of radioactive materials, etc.).

In the operation of a nuclear reactor, the seal of the first containment vessel may be broken, which will also lead to the contamination by radioactive materials of the installation. It is clear that the presence of contamination in air and on surfaces of the

place of work does not exclude the possibility of its penetration inside the human organism. In this case, internal radiation takes place, i.e., the action of ionizing radiation on the organism from radioactive materials, located inside the organism.

In external irradiation by α- or β-particles and γ-photons or neutrons, man is subjected to the adverse effects only while he is /72
located close to the sources of radiation. However, if the radioactive materials enter inside the organism, the individual is subjected to continuous irradiation until the radioactive materials are removed from the organism as a result of decay or physiological exchange. It follows that it is necessary to use various approaches in the evaluation of the hazard to which an individual may be exposed while working with radioactive materials.

Those individuals directly working with radioactive materials and sources of ionizing radiation may not be the only ones subjected to the action of radiation. Various population groups may be subjected to the action of radiation too, since during the technological cycle some amounts of radioactive materials may enter the environment (atmospheric air, water supplies, and soils).

Undoubtedly, the correct organization of the technological process, a closely controlled radiation environment, and the execution of necessary protective measures may decrease the degree of action of radiation on man. However, just as in work in the mine, at home by the kitchen stove, at chemical plants or in flying airplanes, it is impossible to eliminate completely the possibility of an accident and reduce to zero the risk to man, so it is also impossible in the application of atomic energy to reduce to zero the possibility of irradiation.

Whenever radiation materials are used, there always exists a definite probability of disrupting the technological processes, and the occurrence of a radiation accident. Therefore, along with the development of measures for the improvement of technological processes and methods of controlling radiation levels and increasing

the reliability of instrumentation, it is necessary to determine the basis for those radiation levels which will not be dangerous to man and carry a minimum and justifiable risk for the total population in general.

Starting with the presently established facts on the effects of radiation on the living organism, let us examine which principles are available to the respective organizations and agencies responsible for the provision of radiation safety, in the regulation of permissible levels of radiation.

Guidelines for the Standardization of Radiation Levels

In the preceding chapter, it was mentioned that in the estimation of the risk connected with radiation, it is currently assumed that no matter how small the dose, there exists a finite probability of the appearance of genetic and late somatic effects. Moreover, the probability of the risk depends linearly on the radiation dose. These assumptions are used in standardizing radiation levels. /73

Starting from the assumption that no dose is absolutely harmless, according to the recommendations of ICRP, the maximum permissible radiation doses, MPD, are established at levels low enough to eliminate the possibility of acute radiation damage. The risk associated with the appearance of genetic and late somatic effects caused by radiation at these levels must be sufficiently small and justified in comparison with the advantage derived by society from the utilization of artificial sources of radiation. In addition, a further reduction in the risk would not justify those measures of radiation protection which would be required to provide lower levels of radiation [12, 21, 23].

ICRP considers it possible to recommend for individuals working directly with radioactive materials and sources of ionizing radiation, permissible levels of radiation such that the hazard would not exceed that present in an industry or branch of science, where a

high degree of safety is assured. Just as the permissible levels of radiation of various population groups must be such that the risk as a result of using artificial sources of radiation must be less than or equal to the types of risks common in nature (floods, earthquakes, meteorites, etc.), and is justifiable on the basis of the advantages derived from the application of atomic energy [12, 23].

Since the probability of appearance of genetic and late somatic effects depends on the population dose expressed in units of person-rems (see page 72), it is necessary not only to decrease the permissible level of radiation, but to limit the number of persons who may be exposed to radiation. Therefore, it is necessary to avoid such /74
applications of radioactive materials and sources of radiation whose benefits are not apparent. Thus, there is no question about benefits accrued to society from the development of atomic energy, application of radioactive isotopes in various branches of industry and in research. It is impossible to estimate the progress in medicine due to the use of radioactive materials in diagnosis and treatment of a number of serious afflictions. Therefore, society may justify the assumption of a certain degree of risk in the utilization of artificial sources of radiation in the indicated areas of human activity. However, the use of sources of radiation, primarily abroad, for purposes of advertisement and for effect (for example, the use of x-ray machines in the fitting of shoes, and the use of radioactive isotopes in some widely used merchandise — toys, decorations, etc.) appears to be unjustified.

Thus, according to the recommendations of ICRP, the permissible levels of radiation are based on the hypothesis of the lack of a threshold for the action of radiation. In connection with this, numerical values of the maximum permissible dose (MPD) are chosen to be such that the radiation risk at these doses will be minimal, taking into account real economic and social conditions. In practical utilization of radiation sources, one should strive to avoid all unnecessary radiation and to take all sensible precautions which are technically feasible at present, in order that the true radiation

doses be smaller than the prescribed ones by as much as possible.

The established permissible levels of radiation not only define the upper limit of risk that society takes in using artificial sources of radiation, but also serves as a basis for the development of requirements for radiation safety.

It should be kept in mind that the irradiation of various individuals or groups of the population at doses exceeding the permissible ones by relatively small margins (even by several times) will not lead directly to damage to the given group of individuals, since the permissible doses are established at such low levels that the associated risk is significantly less than the risk from natural factors. /75
Therefore, each instance of an insignificant excess in the permissible level of radiation* should be seen in the first place as a sign of the disruption of the normal working conditions in an installation or technical process.

Radiation Safety Standards

Beginning in 1960, permissible levels of radiation were governed by "Sanitary Rules of Work with Radioactive Materials and Sources of Ionizing Radiation SR-333-60 (See Appendix 2, "Maximum Permissible Levels of Ionizing Radiation MPL-1960 [24]).

Associated with the considerable expansion of the application of atomic energy in the national economy and, consequently, with the increase in the group of people who may be exposed to radiation, and in view of the accumulation of new facts dealing with the action of ionizing radiation on the human organism, it became necessary to correct current MPD, and to refine a number of assumptions. Therefore, the Ministry of Health of the USSR developed and introduced, in 1969, "Standards of Radiation Safety NRS-69" [25]. These were developed in accordance with the recommendations of ICRP [12, 21] "Primary Safety Standards in Radiation Protection", prepared by the

*This does not involve increases in radiation levels to values which will pose a threat to the health of individuals (accidents).

International Atomic Energy Agency (IAEA) [13], and the latest studies of Soviet scientists in radiobiology and radiation safety.

A significant difference of NRS-69 from MPL-1960, given in SR-333-60, is the method of establishing the categories of radiation, taking into account possible effects of ionizing radiation on the organism.

The establishment of radiation categories is based on the fact that individuals working in atomic plants or installation, where work with radioactive materials and sources of ionizing radiation is being done, are not the only ones who may be exposed to radiation. Various population groups living in the vicinity of such organizations or institutions may be exposed as well. Radioactive wastes are sources of such an exposure. At the present level of technology, it /76 is impossible to completely eliminate these wastes, as well as those from chemical and steam power plants.

In order to reduce the risk associated with the irradiation of the population living in the vicinity of the plants or installations using atomic energy devices and radioactive substances or sources of ionizing radiation, a sanitary protective zone is established. This is a zone surrounding the plant or installation where living quarters, institutions for children, as well as extraneous industrial and service installations not related to the plant or installation requiring the sanitary zone are prohibited.

For enterprises or installations covering a large territory, the size of the sanitary protective zone may be limited to the property of the enterprise itself.

Beyond the sanitary protective zone is the surveillance zone. This zone includes the territory where radiation doses of the population may exceed the established dose limits.

A systematic control of the radiation environment is required in the surveillance and sanitary protective zones. This makes it possible to estimate the radiation dose to the population living within the surveillance zone, as well as to record operationally

each case of a deteriorating radiation environment, as a result of a disruption of the normal course of the technological process or during an accident, and to take immediate necessary remedial measures. The volume of radiation control (i.e., the number of measurement parameters of the radiation environment and the frequency of control) are determined on the basis of specific conditions.

The dimensions of the sanitary protection and surveillance zones are established only by the principal sanitary-epidemiological agency of the Ministry of Health of the USSR.

In a number of radiation hazardous tasks, the possibility of the entry of radioactive materials into the external medium (for example, in work with gamma-therapy or industrial gamma-installations where sealed sources are used) is completely excluded. It is clear that there is no need in this case to set up sanitary-protective and surveillance zones. /77

According to NRS-69, three radiation categories are established. Category A (personnel) includes individuals who work directly with sources of ionizing radiation or, by the nature of their work, may become exposed to radiation.

Category B (single individuals from the population) includes the population group living in the surveillance zone.

Category C includes the total population, including individuals in Category A and B.

For radiation category A, the maximum permissible somatic doses MPD are established, while for category B, the maximum dose is based on the probability of the occurrence of late effects. The factor limiting the radiation level for category C, which includes the entire population, must be the data on the minimum probability of occurrence of genetic effects. Therefore, in category C, the genetically significant dose is of importance. This dose is defined as the average value of all the gonadal doses received by individuals

in the population, each one of them being weighed by the expected number of children conceived after irradiation.

It should be noted that, according to MPL-1960, the population living in the surveillance zone belonged to category C, and permissible radiation levels were the same as for the entire population. This required the establishment of unjustifiably large sanitary protective zones surrounding atomic plants, and imposed excessively harsh limitations on the amounts of waste. This factor impeded the development of atomic energy, particularly in densely populated regions. The inclusion of individuals living in the surveillance zone in category B, i.e., the establishment of higher permissible levels of radiation for this group than for the rest of the population, does not imply a noticeable increase in the risk of genetic effects for the entire population, while the risk of somatic late effects for this radiation category remains significantly lower than the established level of risk for these effects not caused by radiation.

MPL-1960 established the annual and even weekly dose for category C, which included the entire population. This led to serious difficulties associated with the measurement and evaluation of these doses. At the same time, as previously indicated, the determining factors in the estimation of the risk for the entire population, is associated with the expansion of the use of atomic energy, are the genetic effects. Therefore, the regulation of the genetically significant dose for the entire population permits a correct approach to the determination of the maximum size of the population of a country (for one as large as ours — the size of the population of a republic, region, or province) that can be assigned to radiation categories A and B, in order not to exceed the genetically significant dose. It is clear that this makes it possible to plan a more correct distribution of atomic plants and other installations of the atomic industry. /78

Up to the present time, according to the recommendations of SR-333-60, a personal dosimetric control was required for individuals working directly with radioactive materials and sources of ionizing

radiation. This was justified in the first years of the development of the atomic industry, when the technological processes had not yet firmed up, when there was still insufficient data to estimate the possible variations in the radiation burdens during one process or another. At the present time, on the basis of long-term experience accumulated in the execution of typical operations at atomic installations, such as nuclear reactors, radiochemical plants, and so on, the dynamics of changes in the radiation environment have been rather well studied. On the basis of this experience, expected personnel exposure doses may be estimated. Furthermore, due to the perfection of technological processes, the correct organization of protection, the design of buildings and remote control, working conditions have significantly improved at atomic industrial installations. As a result, radiation doses have been lowered considerably below the maximum permissible levels in a significant portion of individuals occupationally exposed to radiation. Exposures over the permissible levels are practically not observed except in single cases associated with a disruption in an operating procedure or with the necessity of completing emergency repairs. As a result, dose levels for the majority of workers at industrial reactors and power plants do not exceed 50% of the MPD. Furthermore, most of the load is accumulated /79 during maintenance work, when the shields are removed and the reactor core is exposed. The radiation dose has been less than 20% of the MPD during the last decade for more than 80% of the workers [2].

Therefore, for purposes of the organization of dosimetric control and medical observation, two groups were established by NRS-69 for category A.

Group "a", category A, includes individual working in controlled areas, i.e., in buildings or locations of enterprises or institutions where the working conditions are such that the radiation doses may exceed 0.3 MPD. In a controlled area, not only individual dosimetric monitoring is required, but medical observation as well.

Group "b", category A, includes individuals of an enterprise or institution working outside the controlled area, where the expected radiation doses are less than 0.3 MPD. Individual dosimetric monitoring and special medical observation are not conducted for these individuals. It is recommended that the estimate of the possible radiation dose for this category be made by fixed dosimetric equipment, which will also allow the evaluation of changes in the radiation environment.

The principal difference between NRS-69 and MPL-1960 lies in the setting of standards in terms of the maximum permissible intake (MPI) of radioactive materials into the organism, rather than in terms of the previously established maximum permissible concentration (MPC) of radioactive substances in air of places of work, atmospheric air and water. This problem will be considered in greater detail below.

Maximum Permissible Doses

A maximum permissible somatic dose (MPD) for irradiation of the whole body, gonads, or red bone marrow has been set at 5 rem per year for personnel, i.e., individuals working directly with radioactive materials or sources of ionizing radiation, and for those who, by the nature of their work, may be exposed to radiation (category A) [12, 13, 21, 25]. Furthermore, the total radiation dose at N years must not exceed the quantity D determined from the formula:

$$D\ (\text{rem}) = 5\ \text{rem/yr}\ [N\ (\text{yrs}) - 18\ (\text{yrs})]. \qquad (43)$$

The number 18 enters into the formula since, according to our laws, individuals younger than 18 years of age may not be allowed to work with radioactive materials and sources of ionizing radiation. /80

Under continuous irradiation at the maximum permissible dose (5 rem per year) over the entire working life of a person (50 years), no changes will be observed by current methods in the health of the exposed individual and his descendants [25]. In fact, the yearly

group dose for 10^6 occupationally exposed individuals will be $5 \cdot 10^6$ person = rem. The expected number of leukemias per year for such a group will not exceed 5 - 10, i.e., only 10 - 20% of the naturally observed number of leukemias, which is 50 cases per year per 10^6 individuals (see page 70). The probability of occurrence of malignant tumors of other organs and tissues is 10 - 20 cases for a group dose of $5 \cdot 10^6$ person-rem, i.e., only 0.5 - 1% of the natural frequency of observed cases of malignant neoplasm due to non-radiation causes. Thus, irradiation at the level of MPD will bring about a rather insignificant increase in late effects, which practically cannot be detected against the natural background of those effects.

Since the personnel make up a rather insignificant fraction of the total population, the indicated risk is justified for society as a whole, once the advantages of the new form of energy are taken into account. In addition, it should be noted, as mentioned previously, that due to improvements in technology and methods of radiation protection, the radiation level of the majority of workers in the atomic industry are significantly lower than the MPD.

The MPD represents the maximum dose to which personnel may be exposed in the line of duty. It is recommended that work with radioactive substances and ionizing sources of radiation be organized in such a manner that under normal working conditions the actual radiation doses of workers will be as small as possible [12, 13, 25].

Currently available data allow us to state that under chronic, periodic irradiation, the biological effects depend primarily on the total radiation dose received over several years only if the periodic /81 radiation doses are sufficiently low, i.e., lie in the range of permissible levels [12, 21]. Therefore, NRS-69 prescribes a yearly permissible dose, but not a weekly one, as was prescribed by MPL-1960. Furthermore, a periodic and even single dose of 3 rem (i.e., $\sim$ 0.5 yearly MPD) is allowed during 13 consecutive weeks (quarter). However, this dose may be received as a single exposure dose only during a single quarter of the year.

The regulation of the annual dose, as well as the permissible single exposure to a dose of up to 3 rem, allows a more correct organization of work under hazardous radiation conditions, in particular, during maintenance operations, cleanups associated with the disruption of technological processes, etc.

In a number of cases, the situation may arise where the exposure of various workers to doses exceeding the MPD becomes unavoidable. In the first place, all possible measures must be taken to reduce to a minimum the overexposure. In this case, the planned overexposure can be justified only on the basis of extreme necessity, when dealing with human lives, with the prevention of liquidation of the consequences of a radiation accident, associated with radiation effects on a large population group. Personnel must be informed of possible overexposure. In addition, it is necessary to limit the number of individuals who may be subjected to increased exposures. A single dose of 10 rem, i.e., twice the MPD, is permissible in the completion of such tasks. However, additional radiation must be limited in such a way that the total dose D, accumulated during the working life, is less than the value given by the Formula (43) not later than 5 years after the overexposure.

Let us consider the following example. Assume that during the clean-up of the consequences of a radiation accident, the worker was exposed to a dose of D_1 = 10 rem (accumulated dose to the present time D_0 = 65 rem, age N = 32 years). Therefore, after the clean-up of the accident, $D_2 = D_0 + D_1$ = 75 rem. Five years later, i.e., at age 37, the accumulated dose D_3 given by Formula (43) must not exceed $D_3 = 5\ (37 - 18) = 95$ rems. It is clear that the radiation dose accumulated by this worker during the 5 years after the accident cannot be greater than 95 - 75 = 20 rem. Consequently, during the 5 /82
years after the accident, this individual must work under conditions such that his yearly radiation dose will not exceed 4 rem.

In exceptional cases, a single radiation dose of up to 25 rem is allowed. Moreover, this five-fold overexposure with regard to the MPD is allowed only once during the entire working life. This overexposure must be compensated for in such a manner that the accumulated dose will not exceed the value given by Formula (43) 10 years later.

The employment of individuals who have been exposed to a single dose of 0.5 MPD during the preceding 12 months is prohibited in areas of increased radiation exposure. The completion of assignments involving increased preplanned exposures exceeding the MPD by 2 and not more than by 5 times cannot serve as a reason for stopping the worker from fulfilling his usual responsibilities in a radiation area. It is important, however, to create conditions such that the excess exposure will be compensated for after the indicated time lapse.

It is perfectly clear that it is desirable to attract to hazardous radiation work with high exposures, individuals with low accumulated exposures and who may be later used in work involving lower radiation exposures. In all cases of work involving overexposure of individuals of a reproductive age (less than 30 years of age), the accumulated dose by age 30 must not exceed 60 rem, in order to decrease the probability of genetic effects. Based on these same premises, women under 30 are prohibited from work involving high radiation exposures. In general, for women of reproductive age, the total radiation dose during 13 consecutive weeks (quarter) should not exceed 1.3 rem, which corresponds to a yearly dose of 5 rem. Under these conditions, the dose to the embryo during the first two months of pregnancy, when it still cannot be reliably established, will not exceed 1 rem, a safe value [12, 13, 25]. During pregnancy, women are excused from working with unsealed radiation substances during the entire period of breast feedings. This is due to the fact that, /83 in this case, the entry of radioactive material inside the mother's organism is possible, and from her milk — to the baby.

So far, the discussion has dealt with planned increased exposures, when, as a result of circumstances, the management of the enterprise or institution has made the decision to conduct operations when the radiation exposure to certain individuals may exceed the permissible levels. The regulation of possible planned overexposures, prescribed by NRS-69, have been detailed previously.

At the same time, accidents of one type or another are possible at an atomic installation, just as they are at any other industrial plant. As a result of a radiation accident, part of the personnel may be exposed to radiation at dose levels significantly exceeding the maximum permissible ones. It is clear that it is impossible to regulate exposure as a result of an accident, since the accident itself — as well as the related exposure — is an accidental, as well an an extraordinary, event. NRS-69 specifies that any exposure to a dose greater than 25 rem, i.e., exceeding MPD by more than 5 times, must be considered to be potentially dangerous. The worker must be medically examined. If contraindications are not observed, the worker may continue his usual duties. Each overexposure as a result of an accident must be thoroughly investigated, and appropriate measures should be taken.

Taking into account the fact that the potential danger of a radiation accident always exists, the development of a detailed plan is considered essential for every atomic installation. This plan will estimate the degree of the radiation danger resulting from a radiation accident, and will deal with the protection of personnel and, in case of need, with the protection of the population, with an indication of the doses at which one or another measure is to be taken. The main purpose of the measures must be the reduction to a minimum of the radiation burden and the number of exposed individuals [26].

"Temporary Procedural Instruction for the Development of Measures for the Protection of the Population as a Result of Nuclear Reactor Accidents," approved by the Ministry of Public Health of the

USSR, was implemented in 1970. These instructions gave standards for the determination of the radiation environment, for the evaluation of the degree of danger, and for the development of measures for the protection of the population in case of an accident of a nuclear reactor with a water or gas coolant. /84

Individuals not working directly with radioactive substances and sources of radiation may also be subjected to the action of radiation. These include administrative plant personnel who, in the performance of their duties, infrequently enter buildings where radioactive substances and sources of ionizing radiation are handled. Furthermore, individuals working in adjoining buildings, as well as those in all buildings and outdoors within the boundary of the sanitary protective zone, may be exposed. An exposure dose of 0.5 rem per year should be used as a basis for all these individuals in the organization of work with radioactive substances and sources of ionizing radiation, as well as in the protection projections [25].

For individuals living in the control zone and belonging to category B (certain individuals from the population), the control of possible radiation levels may be carried out by monitoring radiation level dose rates in the controlled zone, amounts of radioactive wastes, the radioactivity of objects in the external environment (soil, vegetation, water, air), and corresponding statistical evaluations. Therefore, in contrast with the personnel (category A) for whom the MPD is established and for whom the evaluation of individual radiation doses is anticipated, a maximum dose level is prescribed for specific individuals from the population, based on the average exposure to the population group. Actual doses received by different individuals living in the controlled zone may differ, depending upon age, weight, mass exchange, personal habits, environment, etc. Therefore, in order to estimate the radiation effect on a group, living in the given controlled zone, it is recommended to select the group of individuals (by age, diet, etc.) who are most likely to receive the maximum dose, given by the average exposure dose for the entire population group in the given controlled zone [26].

In contrast with personnel, individuals from the population may include not only adults, but children, as well as pregnant women. As a result, the permissible dose level according to NRS-69 has been set at a very low level, equal to 0.5 rem per year, i.e., 10 times smaller than the MPD established for workers. Exceptions are children, from whom the maximum permissible radiation dose to the thyroid has been established to be 20 times smaller than that of workers. /85

It is natural that the MPD established for workers (category A) is 10 times higher than the dose for individuals from the population (category B), since category A includes a rather small group of people who are constantly under special medical observation, have a reduced work-day and additional leave. Furthermore, individuals younger than 18 years of age, or those having any contraindications as a result of health are not employed in radiation connected work.

The established dose level for separate individuals from the population is only five times higher than the average natural radiation background, and is even lower than the observed maximum value of natural radiation background on Earth, equal to 0.85 rem per year (State of Madras, India) [20]. Therefore, exposure at doses established for category B is associated with a very small risk. In fact, if 10^6 individuals are irradiated to a dose of 0.5 rem, the population dose will be equal to $5 \cdot 10^5$ person · rem, and the expected risk, due to late effects, will be only 0.1 - 0.2 cases of leukemia per year, i.e., 0.1% of the observed natural occurrence of leukemia due to non-radiation factors. Therefore, even with some excess of the established dose limit among groups of individuals of category B, due to natural differences in conditions of life of various individuals in the controlled zone, no threat to the health will be produced, since the degree of risk will remain insignificant [23, 26].

It should be kept in mind that the established values of the MPD and dose limit do not include radiation burdens due to natural radiation background and due to doses received during medical examinations.

The concept of genetically significant dose is introduced to regulate the exposure of the population as a whole (category C). It is the dose which, if received by each individual from the moment of conception to the mean reproductive age, would lead to the same genetic changes in the population as a whole as those produced by actual exposure doses received by various individuals [12, 13, 25].

An analysis of available data, as well as estimates carried out by various international and national organizations, indicates that the genetically significant dose is greater than 6 - 10 rem per person [12, 21]. This means that if the mean gonadal dose from the moment of conception to age 30 (mean reproductive age of man) per person is less than 6 - 10 rem from all sources of radiation produced by man (including radiation from medical procedures, and with the exception of natural background), then the additional damage to society due to genetic consequences may be considered permissible and justifiable, taking into account the expected advantages derived from the development of practical application of atomic energy [12, 21].

According to NRS-69 [25], following the recommendations of ICRP [12, 21], the genetic dose for the entire population (category C) from all sources of radiation, added to that due to natural background and medical procedures, must not exceed 5 rem during 30 years. This value is 50% of the minimal dose, which, according to present day understandings, leads to a doubling of the number of spontaneous mutations [23]. In the estimation of the genetically significant dose, it is necessary to take into account the contribution due to the exposure of personnel (category A), specific individuals from the population (category B), as well as the exposure of the entire population. The latter is associated with the global contamination of the environment as a result of atomic weapons testing, and with other possible uncontrolled sources of radiation — for example, due to an accident involving a space probe using isotope or nuclear sources of energy. Table 7 gives the recommended distribution by NRS-69 of the genetically significant dose according to exposure categories. /87

C-2

TABLE 7. DISTRIBUTION OF THE GENETICALLY SIGNIFICANT DOSE (5 rem per 30 years) ACCORDING TO CATEGORIES OF EXPOSURE [25]

Category of exposure	Genetic dose, rem, over 30 years
Personnel (category A)	1.0
Certain individuals from the population (category B)	0.5
Entire population (category C)	2.0
Reserve	1.5

The assigned genetically significant dose of 1 rem per 30 years to category A implies that personnel may make up 1.7% of the entire population. In addition, they may be exposed to doses of 5 rem per year, starting at age 18. The accumulated dose to age 30 then becomes 60 rem. An assigned genetic dose of 0.5 rem per 30 years to category B means that 3.3% of the entire population may belong to the group of individuals receiving 0.5 rem per year, 15 rem over 30 years, or what is equivalent, 33% of the population exposed to 0.05 rem per year [19, 23, 26].

At the present time, the contribution to the genetic dose by the exposure of workers does not exceed 0.01 rem per 30 years, i.e., 1% of the recommended dose (see Table 7), while the contribution by exposure of the population to radioactive fallout associated with atomic weapons testing and by exposure due to other radiation sources of artificial origin does not exceed 0.2 - 0.3 rem for 30 years [21, 23, 26]. Thus, the total genetic dose from all artificial radiation sources is significantly lower at the present time than the prescribed dose of 5 rem per 30 years and, according to estimates of the ICRP, in all probability will not increase significantly in the near future. As a result of growth in the applications of atomic energy, the genetic exposure dose for the entire population, due to technical radiation sources, will not exceed 1 rem for 30 years [23].

At the same time, the development of atomic energy must provide for a permanent control of the genetically significant dose, and for its maintenance at the lowest possible level by lowering exposure doses of separate groups of individuals, specifically, occupational workers and certain individuals in the population, as well as by limiting the number of individuals who may be exposed.

It should be noted that medical procedures add significantly to the genetic dose. Thus, in countries like France, it is 1.8 rem for 30 years, in Sweden, Japan, and England — 0.8 - 1.2 for 30 years, while in the United States, it is 4.2 rem for 30 years [23]. Therefore, NRS-69 recommends a strict limit on x-ray exposures during mass prophylactic screening of the population, especially those of pregnant women, children, and youths. In carrying out such screenings, the achievement of the medical result should be combined with /88
rational radiation protection, in order to provide a maximum reduction in the genetic dose [25].

Pathways of Entry of Radioactive Substances Inside the Organism

The contamination of the air and working surfaces, and sometimes of special apparel, hands, and the body, may occur in atomic plants or radiochemical laboratories by the disruption of seals of isolation rooms, or pipes, as well as during maintenance work.

Along with this, the contamination of objects in the external environment (atmospheric air, soil, vegetation, water supplies) is not excluded. Potential sources of contamination of objects in the external environment are gaseous, liquid and solid wastes, formed in large quantities at atomic installations. As production technology, transport method, and burial of radioactive wastes become more improved, the significance of this source of contamination decreases. At the present time, solid and liquid radioactive wastes are stored underground or at the surface in special vaults which practically eliminate the leakage of radioactive materials into the external

environment. At the same time, in England and the U.S.A., liquid radioactive wastes, until recently, were disposed of in the seas and oceans. Undoubtedly, this method of waste disposal, in time, would have brought about a significant increase in the activity of the seas.

It is well known that, at the present time, an international agreement has been signed that prohibits the pollution of the seas and oceans by wastes, including radioactive products. This undoubtedly will contribute to the prevention of contamination of objects in the external environment.

It was mentioned previously that an atomic plant accident, as a result of which significant amounts of radioactive wastes may enter the enternal environment, is not unlikely. As a result, atmospheric gases, soils, vegetation, and water supplies will become contaminated by radioactive substances in a given territory around the installation (depending upon the magnitude of the accident).

Radioactive contamination due to wastes from atomic industry or as a result of accidents are of a local nature* which facilitates the implementation of radiation safety measures. /89

In addition, a world-wide source of pollution of the environment exists due to fallout from the upper layers of the atmosphere of radioactive materials accumulated there as a result of atomic weapons testing.

Due to the contamination of the air, surfaces of places of work, and objects of the external environment by radioactive substances, the possibility of their entry into the human organism by various pathways is not excluded. In particular, radioactive materials in the form of gases or aerosols in air, i.e., suspended tiny droplets or solid particles, can penetrate into the lungs in the process of breathing, and are then carried by the blood throughout the organism.

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*We do not consider here disposal at sea.

Radioactive substances may also enter the organism from contaminated surfaces, hands, and special apparel. These pathways of entry will be discussed below (see page 108).

Radioactive materials which settle on the soil are accumulated by the root system of the plants, and may enter the human body by the biological chain: plant — man, or plant — animal milk and meat — man, etc.

Radioactive substances may enter the human organism from contaminated water supplies by the chain: water — algae, plankton, benthos — fish — man; or, if the water supply is potable, they may enter directly by the chain water — man.

The relative number of radioactive substances passing from one link of the biological chain to another is known as a coefficient of discrimination.

It is clear that the fraction of radioactive materials passing from objects of the enternal environment to man are different, while the entry pathways are varied and depend on many factors, including chemical properties of the isotopes, physical-chemical forms of the compounds, in particular — solubility, chemical biological properties of the soil, types of fertilizers, climatic conditions, living conditions, and diet of man in the particular region, etc.

It should be noted that in industrial buildings, the principal /90
pathways of entry of radioactive materials into the organism is by inhalation, i.e., by means of the respiratory system. From objects of the external environment, the pathway is along the food chain, i.e., oral.

Let us consider briefly the fate of the radioactive isotope in the organism as a function of its method of intake.

Radioactive aerosols, depending upon the dispersion of the particles, are retained to a different degree in various parts of

the respiratory system. For example, aerosols with particle size greater than 1 μm are primarily retained in the upper respiratory system. Particles down to 0.1 μm are retained in the tracheobronchial region, while particles smaller than 0.1 μm are retained in the lung alveoli and are then carried to various tissues of the organism by means of the blood [27]. Radioactive isotopes from the upper respiratory tract do not enter other organs and tissues. They are cleared from the organism after a certain time period by various cleansing processes. Thus, the fraction of radioactive materials entering other organs from the lungs depends substantially on the dispersion of the inhaled aerosols. It is also clear that the rate of transfer of radioactive materials from lungs to other systems of the organism is proportional to the solubility of inhaled aerosols in physiologic liquids of the organism, specifically, in the lymph of the blood.

Radioactive substances enter the gastro-intestinal tract by means of oral ingestion. There they are absorbed into the blood and are distributed throughout various organs and tissues. The smaller the solubility of the compound, the greater the fraction that will be transmitted and evacuated from the organism.

It is known that various chemical elements, depending upon their role in exchange processes in the organism, have a tendency to accumulate preferentially in specific organs. For example, radium, strontium, phosphor are primarily concentrated in bone, iodine — in the thyroid, cesium — in muscle, cobalt — in the spleen and in the pancreas, polonium — in the thyroid, kidney; sodium is uniformly distributed in the entire organism, etc.

It is clear that radioactive isotopes of a given chemical element will act in the body in an analogous manner. Consequently, once having entered the blood from the lungs or gastro-intestinal tract, the radioactive isotope will be concentrated preferentially in the organ or group of organs which also concentrate its non-radioactive analog. /91

The retention of one or another chemical element in the organism depends on the intensity with which it takes part in exchange processes. For example, elements such as radium and strontium practically remain in the organism during the entire lifetime. Tritium, cesium, polonium are rather quickly cleared from the organism. The clearance time of an isotope from the body clearly depends on the solubility of the chemical form of the isotope at entry into the organism.

The time required to reduce the amount of a given chemical element in the organism by a factor of two by means of physiological exchange is known as the biological half-life T_b. For radioactive isotopes, the retention time in the organism depends also on the decay half-life. As a result, the concept of the effective half-life is introduced for radioactive isotopes.

The effective half-life T_{eff} is the time required to reduce the amount of a radioactive isotope (activity) in the organism by a factor of two:

$$T_{eff} = \frac{T_F \cdot T_b}{T_F + T_b} \qquad (44)$$

where T_{eff} is the decay half-life of the isotope.

From Formula (44), it follows that if the decay half-life T_f is short, while the biological half-life T_b is large, then T_{eff} will be determined by T_f, and vice versa. /92

Table 8 presents several examples of the values of the effective half-lives for radioactive isotopes.

TABLE 8. EFFECTIVE HALF LIVES OF A FEW RADIOACTIVE ISOTOPES IN GIVEN ORGANS OF RETENTION, DAYS

Isotope	Organ of retention	T_F	T_b	T_{eff}
H^3	Whole body	$4.5 \cdot 10^3$	12	12
Na^{24}	" "	0.63	11	0.6
P^{32}	Bones	14.3	1155	14.3
I^{131}	Thyroid	8	138	7.6
Cs^{137}	Muscle	$1.1 \cdot 10^4$	140	140
Ce^{144}	Whole body	290	563	191
Po^{210}	Spleen	138.4	60	42

Maximum Permissible Doses for Internal Exposure

The biological action of internally deposited radioactive isotopes is determined by the absorbed dose, produced by the radiation, emitted by the given isotope in the process of decay during the time of retention in the organism. Since some isotopes are eliminated exceedingly slowly from the organism, they will subject it to continuous irradiation during a prolonged period of time.

It was previously indicated that various radioactive isotopes are preferentially accumulated in organs and tissues of the organism. However, it is known that the role of various organs in the maintenance of the normal vital activity of the entire organism is different. As a result, the degree of radiation damage from internally deposited radioactive substances in the organism will depend not only on the size of the dose, produced by emitted radiation, but also on the organ, preferentially accumulating the radioactive isotope, i.e., on the critical organ.

The critical organ is the organ in which the accumulation of the radioactive isotope leads to the most severe damage to the entire organism. It should be noted that one and the same isotope, depending upon the solubility of the compound and the pathway of entry into

the organism, may have various critical organs. It is clear that for insoluble compounds, the lungs and the gastro-intestinal tract will be critical organs for inhalation and oral ingestion, respectively. For soluble compounds, on the other hand, the critical organ will be the organ accumulating the chemical element of the given isotope as a result of exchange processes in the organism.

By taking into account the different roles played by various organisms in the sustenance of the vital processes of the organisms, the MPD for external and internal exposures have been established for /93 four groups of critical organs or tissues [12, 13, 25, 28].

Group I. Whole body, gonads, red bone marrow.

Group II. Muscles, fatty tissue, liver, kidneys, thyroid, gastro-intestinal tract, lungs, and other organs, with the exception of those belonging to groups I, III, and IV. The crystalline lens of the eye, which was previously included in group I, is presently included in this group of critical organs. However, for radiations with high LET (neutrons and protons), higher values of the quality factor, indicated in Table 6, are used [12, 13].

Group III. The bony skeleton, thyroid, and skin (excluding the skin of the wrists, forearms, ankle, and foot).

Group IV. Wrist, forearms, ankle, and foot.

Table 9 includes the MPD prescribed by NRS-69 for external and internal exposures for personnel (category A) for various groups of critical organs, and maximum doses for external and internal exposures for certain individuals from the population [25].

It should be noted that for children and youths younger than 16 years of age, maximum dose for the thyroid (Group III of critical organs) must not exceed 1.5 rem per year, i.e., it is twice as small as the established dose for the adult group of certain individuals from the population (category B), which is 3 rem per year.

TABLE 9. MAXIMUM PERMISSIBLE DOSES AND MAXIMUM DOSES FOR VARIOUS GROUPS OF CRITICAL ORGANS OR TISSUES

Group of critical organs or tissues	Maximum permissible dose for workers, rem		Maximum dose for certain individuals of population (rem/yr)
	Per quarter	Per year	
I	3	5	0.5
II	8	15	1.5
III	15	30	3.0
IV	40	75	7.5

In order to estimate the radiation exposure to a given critical /94
organ, it is necessary to know the activity of the isotope, localization in the given organ, type and energy of the emitted radiation, mass and size of the critical organ, as well as the effective half-life of the isotope in the given critical organ. Moreover, if the radioactive decay of the given isotope is not accompanied by γ-radiation, then the entire energy of these particles will be absorbed in the organ of deposition, since the ranges of α- and β-particles in biological tissues are not great. Other organs will not be affected by radiation. However, if the radioactive decay is accompanied by the emission of γ-photons, then in view of their great penetration, other organs and tissues will also be subjected to the action of radiation. However, the exposure dose produced by γ-radiation will be small in comparison with the dose produced by α- and β-particles, and it may be neglected in calculating radiation exposure to the organism. Only in the case of uniform distribution of the isotope throughout the entire organism is it necessary to consider the contribution to the dose by γ-radiation.

If the concentration in the critical organ, C_0 (μCi/g) of the isotope is known, then the exposure dose D (rad), produced in the given organ during a time t (days) from the moment of entry, can be

rather easily calculated from the formula:

$$D = 73\ \bar{E}\ C_0\ T_{eff}\left(1 - e^{-\frac{0.693t}{T_{eff}}}\right) \text{rad.} \qquad (45)$$

Here, $\bar{E}$ is the effective energy of α- and β-particles per disintegration in MeV; T_{eff} is the effective half-life of the isotope in the organism in days.

It is clear that the radiation dose D_∞ to a critical organ for total elimination, i.e., for $t \to \infty$, will be equal to:

$$D_\infty = 73\ \bar{E}\ C_0\ T_{eff} \text{ rad.} \qquad (46)$$

The given formulas apply for uniform distribution of the isotope in the critical organ. For a non-uniform distribution of the isotope, the calculation of the dose becomes more complex.

For a comparative evaluation of the biological effect produced by the entry of the organism of one or another radioactive isotopes, in experiments on animals one can determine what organs or tissues /95
are critical, depending on the pathway of entry and on the physical-chemical properties of the compound. These data make it possible to determine the maximum permissible content (MPC) of the radioactive isotope in the critical organ for which the radiation dose will become equal to the maximum permissible one in accordance with Table 9.

To illustrate this, Table 10 gives values of the MPC for a few radioactive isotopes, i.e., activities which will produce the maximum permissible radiation burden in the critical organ (category A).

Data in Table 10 indicate the following: if the activity of the isotope in the critical organ will remain constant (as a result of continuous entry, at a rate equal to its elimination from the critical organ), then the radiation burden of the critical organ will be /96
equal to the maximum permissible one. For example, if, during the

TABLE 10. MAXIMUM PERMISSIBLE AMOUNTS OF RADIOACTIVE ISOTOPES IN ORGANS [13, 25]

	Form of isotope in compound	Critical organ	MPA, μCi
C^{14}	Soluble	Fat	$1.6 \cdot 10^2$
P_{32}	"	Bone	3.1
	Insoluble	Lungs	1.2
Co^{60}	Soluble	Whole body	13
	Insoluble	Lungs	1.2
Sr^{90}	Soluble	Bone	2.0
	Insoluble	Lungs	0.76
I^{131}	Soluble	Thyroid	0.15
	Insoluble	Lungs	2.8
Cs^{137}	Soluble	Muscle	14
	Insoluble	Lungs	2.0
Ir^{192}	Soluble	Kidneys	0.5
	Insoluble	Lungs	1.4
Po^{210}	Soluble	Spleen	0.0020
	"	Kidneys	0.0045
	Insoluble	Lungs	0.015
Ra^{226}	Soluble	Bone	0.10
	Insoluble	Lungs	0.0152
Pu^{239}	Soluble	Bone	0.041
	Insoluble	Lungs	0.016

year, the amount of iodine in the thyroid remains 0.15 μCi, while that of Cs^{137} in muscle is equal to 14 μCi, then the radiation dose will be equal to the MPD, i.e., the radiation dose to the thyroid will be 30 rem per year, and 15 rem per year to the muscle (see Table 9).

Maximum Permissible Intake of Radioactive Substances

Working conditions should be created such that, if the entry of radioactive substances into the organism is possible, the annual MPD prescribed by NRS-69 for the critical organ (see Table 9) will not be exceeded. Consequently, the entry of radioactive substances through the respiratory of digestive systems must be such that the amount of the radioactive isotope in the critical organs will not exceed the MPA, i.e., the value corresponding to the maximum permissible radiation burden. It was previously mentioned that in industry, the most probable and principal pathway of intake of radioactive substances into the organism is by inhalation. As a result, NRS-69 established for personnel (category A) an annual MPI (maximum permissible intake) of radioactive substances through the respiratory system.

Since the contamination of the air, water supplies, soils, and vegetation by atomic industries in the area may be caused by radioactive wastes or as a result of a radiation accident, maximum annual intakes (MAI) of radioactive materials through the respiratory and digestive systems have been established for certain individuals from the population (category B).

The prescribed MAI serve as a basis for the establishment of the zone of explosions and permissible wastes of radioactive gases and aerosols from the stacks of atomic power plants or radiochemical laboratories, as well as for determining the possibility and magnitude of utilization of contaminated water, fish, vegetables, grasses, or other food products in case of need.

Table 11 gives some examples of the MPI and MAI for certain isotopes.

In accordance with NRS-69, a single or short-term intake of radioactive isotopes into the organism (equal to half of the MPI) is allowed, provided that the actual annual intakes do not exceed

TABLE 25. MPI AND MAI OF SOME RADIOACTIVE ISOTOPES [25]

	Form of isotope in compound	MPI for personnel, respiratory system, μCi/yr	MAI for certain individuals from population, μCi/yr	
			Respiratory system	Digestive system
C^{14}	Soluble	$8.7 \cdot 10^3$	$8.7 \cdot 10^2$	$6.6 \cdot 10^2$
P^{32}	Soluble	$1.8 \cdot 10^2$	18	15
	Insoluble	$2.0 \cdot 10^2$	20	18
Co^{60}	Soluble	$8.0 \cdot 10^2$	80	39
	Insoluble	22	2.2	28
Sr^{90}	Soluble	2.9	0.29	0.32
	Insoluble	14	1.4	28
I^{131}	Soluble	21	2.1	1.6
	Insoluble	$8 \cdot 10^2$	80	51
Cs^{137}	Soluble	$1.6 \cdot 10^2$	16	12
	Insoluble	36	3.6	35
Tm^{170}	Soluble	87	8.7	37
	Insoluble	87	8.7	37
Ir^{192}	Soluble	$3.1 \cdot 10^2$	31	32
	Insoluble	64	6.4	30
Po^{210}	Soluble	1.2	0.12	0.58
	Insoluble	0.5	0.05	23
Ra^{226}	Soluble	$7.1 \cdot 10^{-1}$	$7.1 \cdot 10^{-2}$	$9.6 \cdot 10^{-2}$
	Insoluble	0.2	0.02	26
Pu^{239}	Soluble	$4.3 \cdot 10^{-3}$	$4.3 \cdot 10^{-4}$	3.6
	Insoluble	$9.5 \cdot 10^{-2}$	$9.5 \cdot 10^{-3}$	23

the MPI. Just as in internal exposure, in exceptional cases, the intake of radioactive materials is allowed to exceed the MPI by a factor of two in each single instance, or by a factor of 5 over the entire period of work. However, this can be justified only when it is impossible to take measures which will prevent such overexposure of personnel from occurring. Moreover, in each case, the worker must be warned about the possibility of the intake of radioactive materials exceeding the MPI inside the organism. /97

The primary new feature in NRS-69 in comparison with the previous standards [24] are the regulations dealing with the recommendation of the ICRP, the maximum permissible annual intakes (MPI) of radioactive isotopes in air for breathing during working hours, and /98
maximum annual intake MAI through the respiratory and digestive systems for single individuals from the population. The inclusion of the MPI and MAI is undoubtedly a step forward on the road to a more objective evaluation of the danger of internal exposure, to improvement in the control of radiation, and to a more rational organization of work under conditions of possible intake of radioactive substances inside the organism.

According to NRS-69, the annual intake of radioactive materials, rather than their concentration in air or water, turns out to be the most significant parameter in the evaluation of the danger from internal exposure. The use of concentrations as standards assumed that exceeding the MPC by any amount, even for a short term, was inadmissible, even though the amounts of radioactive materials in the medium of interest may have been, moments before or after, significantly lower than the MPC, compensating fully for the excess.

Furthermore, investigations of the radiation environment show that, as a rule, the concentration of radioactive materials, measured at various points of a building, vary considerably depending upon type and size of the building, location and nature of the sources of radioactive materials, disruptions of the ventilation systems, their power, and other parameters which are difficult to take into account. It is practically impossible to evaluate the uptake of radioactive materials by the organisms of specific workers on the basis of measurements of concentrations at one or even at a few points in a building at various different intervals of time. Consequently, it is impossible to evaluate the radiation burden, due to the uptake of radioactive materials by the organism. Therefore, in evaluating the hazard due to internal exposure and in making decisions on the organization of work, the constraint is the intake or content of the isotope in the human body, rather than the concentration in the surrounding medium.

However, on the basis of a greater number of measurements of the instantaneous concentration during a prolonged time interval (for example, a year), it is possible to determine the average annual concentration of a radioactive material in the air of a given work building. The knowledge of the average annual concentration, as well as the duration of the worker's stay in that building, will make it possible to estimate approximately the annual uptake of radioactive materials inside the organism. Furthermore, it is possible to evaluate the disruption of the technological process and to take necessary measures to prevent the overexposure of personnel on the basis of data on the concentration of radioactive materials in the air of work buildings. /99

In order to carry out conveniently the operational control over the radioactive content of materials in the environment, mean annual permissible concentrations (MAPC) were established for radioactive materials in the atmosphere of work buildings, as well as in the atmosphere outdoors and in water for certain individuals from the population. Moreover, short term excursions exceeding the MAPC in the environment should not be considered as being dangerous, since mean annual concentrations, rather than instantaneous values (as prescribed by SR-333-60 [24]) are presently being regulated. In this connection, one of the important objectives of radiation protection services is the compilation of the necessary statistical data on the measurement of instantaneous concentrations for a more reliable determination of the mean annual concentrations in various work buildings, as well as in atmospheric air surrounding the installation, and on the accurate record of the time spent by various individuals in one building or another.

Cases of single or repeated excursions above the MAPC, carrying a potential danger of exceeding the MPI of radioactive isotopes for corresponding categories of irradiated individuals, should be investigated and their causes eliminated [28].

The following relationship exists between the MPI and MAPC:

$$\text{MPI } (\mu\text{Ci/yr}) = 10^6 \text{ MAPC } (\text{Ci}/\ell) \cdot Q\ (\ell/\text{yr}), \qquad (47)$$

where Q is the amount of inhaled air or consumed water per year.

It is assumed that the volume of air inhaled by personnel (category A) during working hours (36-hour work week) is 2500 m^3/yr ($2.5 \cdot 10^6$ ℓ/yr), while that inhaled by an adult individual from the population is 7300 m^3/yr ($7.3 \cdot 10^6$ ℓ/yr). The consumption of water by an adult individual in food products or in liquid form is 800 ℓ/yr (2.2 ℓ/day) [12, 13, 21, 25].

Examples of values of the MAPC are given in Table 12 for several radioactive isotopes.

TABLE 12. MEAN ANNUAL MAXIMUM PERMISSIBLE CONCENTRATIONS, Ci/ℓ, FOR CERTAIN RADIOACTIVE ISOTOPES [25]*

Radioactive isotope	For all personnel in the air of work buildings	For certain individuals from pop.	
		In air	In water
C^{14}	$3{,}5 \cdot 10^{-9}$	$1{,}2 \cdot 10^{-10}$	$8{,}2 \cdot 10^{-7}$
P^{32}	$7{,}2 \cdot 10^{-11}$	$2{,}4 \cdot 10^{-12}$	$1{,}9 \cdot 10^{-8}$
Co^{60}	$8{,}8 \cdot 10^{-12}$	$3 \cdot 10^{-13}$	$3{,}5 \cdot 10^{-8}$
Sr^{90}	$1{,}2 \cdot 10^{-12}$	$4 \cdot 10^{-14}$	$4 \cdot 10^{-10}$
I^{131}	$8{,}4 \cdot 10^{-12}$	$2{,}9 \cdot 10^{-13}$	$2{,}0 \cdot 10^{-9}$
Cs^{137}	$1{,}4 \cdot 10^{-11}$	$4{,}9 \cdot 10^{-13}$	$1{,}5 \cdot 10^{-8}$
Tm^{170}	$3{,}5 \cdot 10^{-11}$	$1{,}2 \cdot 10^{-12}$	$4{,}6 \cdot 10^{-8}$
Ir^{192}	$2{,}6 \cdot 10^{-11}$	$8{,}7 \cdot 10^{-13}$	$3{,}7 \cdot 10^{-8}$
Po^{210}	$2{,}0 \cdot 10^{-13}$	$6{,}8 \cdot 10^{-15}$	$7{,}3 \cdot 10^{-10}$
Ra^{226}	$1{,}0 \cdot 10^{-13}$	$3{,}5 \cdot 10^{-15}$	$1{,}2 \cdot 10^{-10}$
Pu^{239}	$1{,}7 \cdot 10^{-15}$	$5{,}9 \cdot 10^{-17}$	$4{,}5 \cdot 10^{-9}$

*Translator's note. Commas in numbers represent decimal points.

In conclusion of the present section, it should be noted that /100 the MPI and MAI are not regulated for the radioactive noble gases argon, krypton, xenon, and for the short-lived isotopes of carbon, nitrogen and oxygen, since the radiation hazard associated with their presence in air is determined not by internal, but by external exposure, as a result of their rapid elimination from the organism.

Therefore, the radiation danger resulting from an atmosphere containing these isotopes is determined by the dose of external β- and γ-irradiation.

In most cases, water and air may be contaminated not by one, but by a number of various isotopes, for example, by a mixture of uranium fission products. In this case, the MPI on water and air may be determined from the formula:

$$MPI_{mix} = \frac{\sum MPI_i}{\sum \frac{p_i}{MPI_i}}, \quad (48)$$

$$MAPC_{mix} = \frac{\sum MAPC_i}{\sum \frac{p_i}{MAPC_i}}, \quad (49)$$

where MPI_i is the maximum permissible intake of the i^{th} isotope in the mixture; p_i is the percentage contribution of the i^{th} isotope (separately for α- and β-emitters); $MAPC_i$ is the mean annual maximum permissible concentration of the i^{th} isotope in the mixture. If the relative contributions of each isotope in the mixture are unknown, then the MAPC of the mixture is determined on the basis of the most toxic isotope. /101

For mixtures of isotopes of unknown composition, the following values of the MPI and MAPC in water and air are assumed (Table 13).

It should be noted that for purposes of comparison of the hazard due to possible internal exposures in working with unsealed radioactive materials, all radioactive isotopes are divided into five groups according to their radiotoxicity [28].

The assignment to a particular group of toxicity is made on the basis of the maximum permissible activities of the radioisotopic sources at the place of work, not requiring licensing or permission from the sanitary epidemiological services. These values of the maximum permissible activities of the sources at the place of work are given in NRS-69 [25].

TABLE 13. MAXIMUM PERMISSIBLE INTAKE AND MEAN ANNUAL PERMISSIBLE CONCENTRATIONS FOR ISOTOPIC MIXTURES OF AN UNKNOWN COMPOSITION*

Composition of mixture of unknown constitution	MPI for personnel, respiratory system, μCi/yr	MAI for certain individuals from pupulation, μCi/yr		MAPC, Ci/l		
		Respiratory	Digestive	Air of work bldgs.	Atmospheric air	Open water supplies
α-emitter	0,001	0,0001	0,08	$4 \cdot 10^{-16}$	$1,4 \cdot 10^{-17}$	$1,0 \cdot 10^{-9}$
β-emitter (no Ac^{227} β-emitter)	0,06	0,006	—	$2,4 \cdot 10^{-14}$	$8,4 \cdot 10^{-16}$	—

*Translator's note. Commas in numbers represent decimal points.

Elements of especially high radiotoxicity — Group A. These include isotopes whose maximum permissible activity at the place of work MPA is equal to 0.1 μCi. For instance, Pu^{239}, Po^{210}, Ra^{226}, and others. /102

Elements of a high radiotoxicity — Group B. These include isotopes with MPA equal to 1 μCi. For instance, Ra^{223}, Sr^{90}, I^{131}, and others.

Elements of moderate radiotoxicity — Group C. These include isotopes with MPA equal to 10 μCi. For instance, Pm^{147}, Ir^{192}, Cs^{137}, and others.

Elements of least radiotoxicity — Group D. These include isotopes with MPA equal to 100 μCi. For instance, C^{14}, Be^{7}, Zr^{97}.

Elements with MPA equal to 1000 μCi belong to Group E. For instance, H^{3}.

It is clear that such a distribution of isotopes into groups according to radiotoxicity is made for the convenience in evaluating hazards of working with one or another isotope, and is of an arbitrary nature. At the same time, it allows one to carry on work with certain amounts of radioactive isotopes in conventional installations, without the necessity of obtaining permission from the sanitary-epidemiological services for this work. For instance, it is possible to work with 0.1 μCi of Po^{210}, 1 μCi of Sr^{90}, 10 μCi Cs^{137}, 100 μCi Zr^{97}, and 1000 μCi H^3 in conventional laboratory installations without special permission.

Permissible Levels of Contamination by Radioactive Materials of Working Surfaces and of the Body

In working with radioactive materials, in particular, in maintenance work or in the clean-up of spills or leakages of radioactive materials from technological systems (for instance, systems of the primary shield of the nuclear reactor), the contamination of working surfaces, and sometimes of the hands and bodies of the workers, by radioactive materials becomes possible. In this case, the radioactive substances present a hazard as sources of external irradiation in the contamination of hands and body. In addition, the contamination of the surfaces and body may serve as potential sources of internal exposure for two reasons. In the first place, the radioactive materials contaminating the floor, walls, or equipment in a building could be blown off into the air, due to the movement of people and /103
the execution of various tasks, creating increased concentrations of aerosols. In the second place, the radioactive materials may enter the organism as a result of being absorbed through the contaminated skin. Furthermore, the possibility cannot be eliminated that the radioactive substances can be hand-carried to the mouth.

Even though the radiation burden to the skin can be rather accurately estimated and, consequently, the permissible contamination levels determined on the basis of only the external interaction factor, it is significantly more difficult to estimate internal exposure

factors, which depend on many parameters. In fact, the fraction of radioactive materials which may enter the organism as a result of skin absorption depends on the skin condition of a given individual, on the physical-chemical properties of materials on the skin, on the humidity and temperature of the air in the building, nature of the work, etc. [29].

The fraction of the radioactive materials passing from contaminated surfaces to the air of the work building (coefficient of transfer) also depends on many factors: on the nature and intensity of the work in the building, on the amount of equipment in the building, on the material of the contaminated surface, and on the physical-chemical properties of the radioactive materials contaminating the working surfaces, multiple exchanges of air, etc. An even greater uncertainty and difficulty lies in estimating the coefficient of transport of the radioactive materials into the organism through the mouth from contaminated hands. This coefficient, in a certain sense, is mainly subjective. For example, let us note that the fraction of plutonium absorbed through the skin varies from $1.8 \cdot 10^{-3}$ to $5 \cdot 10^{-5}$ (0.18 - 0.005%) [30]. For the most radiotoxic element Sr^{90}, this coefficient is $3 \cdot 10^{-2} - 4 \cdot 10^{-3}$ (3 - 0.4%) [31]. The coefficient of transfer varies between 10^{-5} to 10^{-3} [32, 33], i.e., approximately one thousandth to a tenth of one percent of the activity is transferred to a unit volume from a unit area. The coefficients of transport of radioactive materials to the gastrointestinal tract through the mouth by means of contaminated hands is set arbitrarily at not more than 10%.

In view of the large uncertainty in the values of the coefficient of transfer of radioactive materials from contaminated surface to air and into the organism, as well as in the values of the coefficient of absorption through the skin, the most unfavorable conditions, i.e., the maximum values of the coefficient of transfer were taken in the establishment of the permissible levels of contamination of surfaces and the body. Furthermore, these standards /104

TABLE 14. PERMISSIBLE LEVELS OF CONTAMINATION OF THE SKIN (particles/cm^2 · min), PERSONNEL AND SURFACES OF WORK BUILDINGS IN THE CONTROLLED AREA [25]

Contaminated object	α-emitting isotope		β-emitting isotope
	High toxicity	Other	
Skin of personnel (category A)	5	5	100
Surfaces of work buildings and equipment:			
a) Permanently assigned personnel	10	40	2000
b) Partially serviced (building in a three zone design)	100	400	8000

are significantly on the safe side, since methods of protecting the respiratory system are mandatory in the performance of work in a contaminated building. The work is conducted in special apparel and gloves, a fact which significantly reduces the probability of a contaminated body (see Chapter 4).

Tables 14 and 15 present the permissible levels of contamination of external surfaces of equipment, instruments, the surfaces of work buildings where radioactive materials are used, as well as of skin surfaces of personnel and external surfaces of individual protective agent. Specific permissible levels of contamination have been established for all β-emitting isotopes, and for two categories of α-emitting isotopes: highly toxic, and all others. The highly toxic α-emitting isotopes include those with a MPC in the air of work buildings less than $2 \cdot 10^{-15}$ Ci/l — for instance, Pu^{239}, Cf^{251}, and a few other transuranium α-emitting isotopes.

Clearly, the level of contamination of surfaces and hands must /105
be established only for personnel, since no amount of contamination is permissible in locations where work with radioactive substances is not conducted.

TABLE 15. PERMISSIBLE LEVELS OF CONTAMINATION BY RADIOACTIVE MATERIALS OF PERSONNEL PROTECTIVE ARTICLES (particles/cm^2 · min) [28]

Contaminated object	α-emitting isotope		β-emitting isotope
	High toxicity	Other	
Towel	5	5	100
Special apparel	5	5	100
Inner surface of facial portion of protective article	5	5	100
Basic work clothes	10	40	800
Additional protective clothing;			
a) Inside surfaces	10	40	800
b) Outside surfaces	100	400	8000

Levels of radioactive contamination of internal surfaces of chambers, isolation rooms, exhaust hood, as well as equipment located there, are not regulated. Furthermore, the contaminated objects and surfaces must not contaminate the air at levels greater than the established MPC in work buildings, or bring about an overexposure of hands working in exhaust hoods and chambers [28].

In contrast with the "Sanitary Rules" implemented in 1960 (SR-333-60) [24], the "Principal Sanitary Rules" approved in 1972 (PSR-72) [28] do not regulate the levels of contamination of hands and special apparel prior to and after cleaning, since this does not have any special practical significance.

The skin should be decontaminated down to background values. Clothing and special apparel should also be decontaminated down to minimum possible levels. However, when it is impossible to achieve this, it is allowed to use special apparel with a level of contamination not exceeding values given in Table 15. Contamination of personal clothing and shoes is not permissible. It should be noted that if the activity cannot be removed from the surface (i.e.,

cannot be decontaminated by the usual methods) then it presents a lesser hazard than if it could (for example, unwashed special apparel). The transfer of activity to the air and to the body and, consequently, its intake into the organism is less probable in the first case than in the second. /106

In much the same way, the differential approach to the establishment of contamination levels for surfaces for various buildings appears to be fully justified, considering that these standards are established with a large margin of safety. Actually, in the second zone (see Chapter 6) where maintenance workers are provided with special apparel, additional methods of individual protection and respirators, and where a sanitary workup is always made, there is no need to regulate the levels of contamination as stringently as in buildings housing, for example, conventional radiochemical laboratories of the II and III classes where the spread of materials beyond the confines of the place of work is more likely.

Moreover, a less stringent regulation of the contamination levels in the first case and the strict control at the exits from the zone eliminates, on one hand, the possibility of the spread of radioactive materials beyond the boundary of the zone, and at the same time improves the capability of conducting maintenance work.

CHAPTER 5

CERTAIN ASPECTS OF THE DOSIMETRY OF IONIZING RADIATION

Methods of Detecting Radiation

Instrumentation capable of detecting the radiation level is necessary to evaluate the radiation environment at locations where work with radioactive materials is conducted. The principles of operation of any instrument designed for the detection of penetrating radiation is based on the measurement of the effects produced during the interaction of radiation with matter.

The ionization detection method is the most widely used one up to the present time. It is based on the measurement of the direct effect of the interaction of radiation with matter, i.e., on the degree of ionization of the medium traversed by the radiation. /107 Other methods of detection are based on the measurement of secondary effects, produced by ionization (darkening of photographic film, luminescence of certain materials under the action of radiation, changes in the chemical or physical properties of the materials.

Let us examine in greater detail the methods of detecting ionizing radiation, which currently find practical application.

Ionization method

Figure 19 gives the circuit for an ionization chamber. It includes an air capacitor with two charged plates A and C, with an applied voltage produced by the battery B. In the absence of a radiation source, the air acts as an insulator and no current flows through the capacitor. If ionizing radiation interacts with the air layer between the electrodes, ions will be produced in the gas. These ions will move under the action of the electric field toward

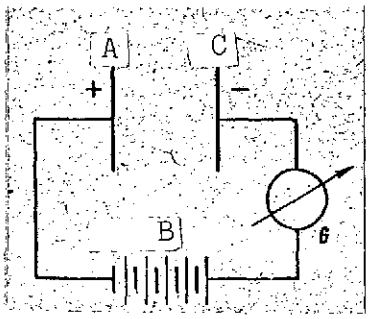

Figure 19. Circuit diagram of an ionization chamber

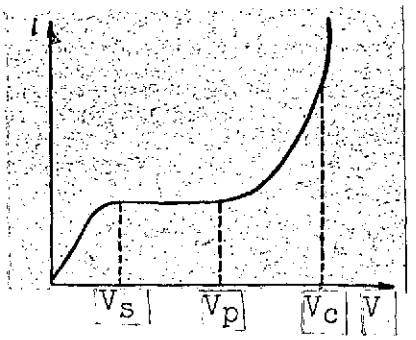

Figure 20. Volt-ampere characteristic curve for an air capacitor

the plates and the meter of the galvanometer G will indicate the flow of an electric current in the circuit.

Figure 20 shows the dependence of the intensity of the ionization current i on the voltage V, applied to the plates of the capacitor. This curve is known as the volt-ampere characteristic curve of the air capacitor. If the voltage is not large, then a fraction of the ions will be able to recombine during the movement towards the charged plates. Increasing the voltage decreases the probability /108 of recombination. As a result, the intensity of the ionization current increases.

At some initial voltage V_s, all ions formed by radiation will reach the plates of the capacitor, and a further increase in voltage will not increase the ionization current. The voltage range from V_s to V_p, over which the ionization current remains constant, is known as the saturation region. The intensity of the ionization current passing through the air capacitor at these voltages is known as the saturation current. Having measured the magnitude of the ionization current i in the saturation region, it is easy to determine the number of ion pairs n_0 formed per second by radiation, and thus indirectly to determine the dose rate of the radiation as well:

$$i = n_0 e, \qquad (50)$$

where e is the charge on the electron ($e = 4.8 \cdot 10^{-10}$ CGSE = $1.6 \cdot 10^{-19}$).

If the voltage exceeds V_p, the ionization current will once again begin to increase, since the electrons formed by radiation in the process of ionization will gain enough energy in their movement towards the positively charged plate to produce ionization by themselves. This process is known as ionization by collision. The energy gained by electrons is proportional to the voltage as is the number of ion pairs formed by them in the process of gas amplification.

In the voltage range from V_p to V_c, there is a strict proportionality between the number of ion pairs n_0 formed per second by the primary radiation and the total number of ion pairs n formed per second in the volume of the gas:

$$n = kn_0; \tag{51}$$

where k is a constant for a given voltage which is known as the gas amplification factor. This factor gives the ratio of the total number of ion pairs produced in the gas space due to the collision ionization to the number produced by the primary radiation. Since the intensity of the ionization current is equal to the product of the total number of ion pairs n in the gas space and the charge on the individual ion e, we obtain from Equation (51):

$$i = kn_0e. \tag{52}$$

Thus, it is possible to determine in this case the number of ion pairs formed and, consequently, the dose rate of radiation as well, by measuring the ionization current i and by knowing the gas amplification factor k. /109

Let us consider further the volt-ampere characteristic given in Figure 20. If the voltage between plates will become greater than V_c, then the production of a single ion pair in the gas space

will give rise to a current between the plates. Furthermore, its intensity is independent of the initial ionization. This region of the volt-ampere characteristic curve is known as the continuous discharge.

In this case, even if only one ionizing particle passes through the capacitor, it will produce a discharge between the plate, and therefore will be detected.

Ionization chambers. An air or gas capacitor operating in the saturation region is known as an ionization chamber, provided that it has an applied voltage between V_s and V_p (see Figure 20) and is designed for the measurement of ionization produced by radiation.

The saturation voltage of ionization chambers operating at normal pressures is 100 - 300 V.

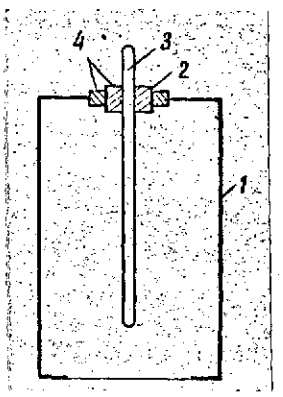

Figure 21. Diagram of an ionization chamber

Ionization chambers are most frequently cylindrical in shape (Figure 21). The body 1 of the chamber is grounded, while the rod 3, known as the central electrode, is charged positively. To decrease the leakage between the central electrode and the body, a guard ring 2 is introduced, mounted on large resistance insulators 4 (amber, polystyrene, etc.). The guard ring is approximately at the same potential as the central electrode.

The ionization currents produced in the chamber, even with exposure to large doses, are extremely small (on the order of 10^{-12} - 10^{-11} ampere). For example, with a dose rate of 100 μR/sec (125 times greater than the maximum permissible dose rate), the ionization current in a chamber 1 liter in volume will be approximately $3.3 \cdot 10^{-11}$ amperes. Such currents cannot be measured directly by the /110

usual instrumentation. They must first be amplified by special radio engineering methods. Since the ionization produced by radiation in a given volume may be measured by means of ionization chambers, they are commonly used to measure the dose rates or doses of various types of radiation.

To measure the dose produced by the flux, for example, of α-particles, it is necessary that the entire range of the particles be included in the volume of the chamber. In this case, the entire energy of the α-particle will be expended in the production of ionization in the chamber. The dose produced by the source of the α-particles may be determined from the ionization current. In order to eliminate the absorption of α-particles in the walls of the chamber, very thin windows are used to permit the entry of α-particles into the ionization chamber, or the α-active substances being measured are placed inside the chamber.

It is clear that it is practically impossible to measure with ionization chambers the dose produced by β-particles, since the ranges in air of the β-particles are several meters long. Fluxes of β-particles are more frequently measured by means of gas-discharge counters.

The measurement of gamma radiation is made by means of chambers such that the ionization in the air volume is caused by secondary electrons ejected from the walls. If the wall thickness is equal to or greater than the maximum range of the secondary electrons (in particular, the condition of electron equilibrium exists, see Chapter 3), then having measured the ionization in the air cavity, i.e., having measured the exposure dose, we can determine the absorption of energy of the radiation in the walls of the chamber. In order to determine the absorbed dose in biological tissues from the measured ionization, it is necessary to make the chamber wall from a material with an atomic number close to that of biological tissue. Plexiglass, resite, and polystyrene serve this purpose. The inside surfaces of the chamber walls, made from insulating materials, are coated with conducting aquadag.

The number of ion pairs formed by the secondary particles in the air volume of the chamber is proportional to the volume of the chamber. Therefore, relatively large volume chambers, 0.5 - 5 ℓ, /111 are used in the measurement of small doses of radiation. Small chambers, a few cubic centimeters or even fractions of a cubic centimeter in volume, are used in the measurement of large doses.

The ionization chamber may be used to detect thermal neutrons. The chamber must be coated with cadmium. It is known that a layer of cadmium 0.5 - 1 mm thick will almost completely absorb a flux of thermal neutrons of any intensity. Gamma-photons are emitted as a result of thermal neutron capture by the nuclei of cadmium.

After having properly calibrated the chamber, it is possible to determine the dose or flux of thermal neutrons on the basis of the dose produced by γ-radiation.

In order to measure the absorbed dose of fast neutrons, it is necessary to select the wall material of the ionization chamber in such a fashion that the atomic composition is similar to that of biological tissue. The absorbed dose is measured by means of the ionization produced in the air cavity by the recoil nuclei formed in the chamber walls by the neutrons.

Proportional counters. If the voltage applied to the electrodes of the chamber is in the range V_p - V_c (see Figure 20), the intensity of the ionization current will increase as a result of collision ionization. As mentioned previously, there is a strict proportionality in this voltage range between the number of ion pairs formed by radiation and the intensity of the ionization current. Ionization instruments operating in this voltage range are known as proportional counters. The gas amplification factor k is usually between 1 to 10,000. This makes it possible to measure less intense radiation.

Proportional counters are usually cylindrical or flat. A thin metallic wire serves as a positive electrode.

These counters are most frequently used in the detection of α-radiation.

Geiger-Mueller counters. Ionization instruments operating in the continuous discharge region, i.e., for $V > V_c$ (see Figure 20), are known as Geiger-Mueller counters. They are most frequently used to detect β-particles. They consist of a conducting cylinder and a thin metallic wire 0.1 - 0.2 mm thick, insulated from the cylinder. The counters, based on the design, are divided into cylindrical (Figure 22) and end-window (Figure 23) types. One of the ends of /112

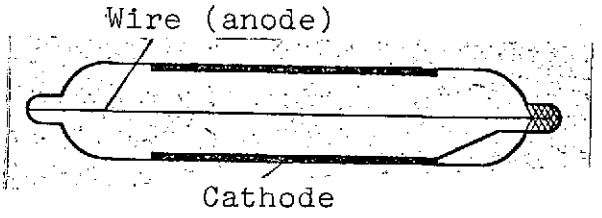

Figure 22. Diagram of a cylindrical Geiger-Mueller counter

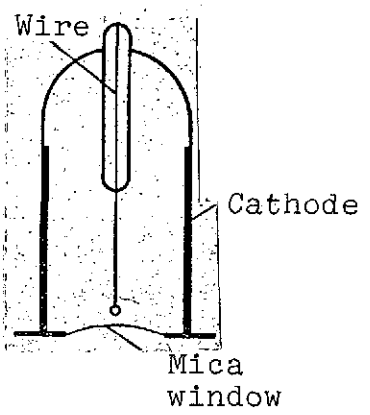

Figure 23. Diagram of an end-window counter

the wire in the end-window counter is unsecured, and terminates in a small glass ball. One end of the counter is covered with a thin mica window, not thicker than 5 mg/cm^2, in order to permit the entry into the chamber of low energy β-particles. A positive potential on the order of 800 - 1000 V is applied to the wire, while the cylinder is grounded. The counter is filled with gas at reduced pressure (50 - 600 mm Hg). Counters filled with special gases (halogen counters) operate at lower potentials (on the order of 300 - 400 V).

Let us examine the discharge mechanism in the counter operating in the continuous discharge region.

Let us assume that the charged particle (α or β) forms at least one ion pair in the volume of the counter. As the electron approaches the wire, where the electric field potential increases greatly, it

will be accelerated and will form a large number of secondary particles as a result of collisions. The accelerated secondary electrons will, in turn, produce ionization. As a result, the number of electrons moving towards the wire will increase exponentially. Therefore, on the order of 10^9 ion pairs are formed after a short period of time (approximately 10^{-7} seconds). This leads to the development of a self-sustaining gas discharge in the counter, since positive ions will, in turn, eject additional electrons on reaching the cathode (cylinder of the counter). The latter will produce ionization /113
by collision in moving to the wire, forming a new avalanche of electrons. Furthermore, electrons in the volume of the counter will also produce in the gas discharge, as a result of the ejection of electrons from the cathode, ultraviolet radiation by the transition of atoms from excited to ground states.

During the gas discharge, the counter is unable to detect ionizing particles passing through it. Therefore, in order to quench the discharge in the counter, a large resistance, on the order of 1000 MOhm is connected in series with the counter. Since the discharge produces an electrical current in the counter circuit and extremely large voltage drop is produced across the resistance, the continuous discharge is interrupted and the potential applied to the wire then returns to its original value. After this occurs, the counter is able to detect the next particle passing through it. The quenching mechanism terminates the discharge in 10^{-4} - 10^{-2} seconds. This implies that the counter is able to detect 10,000 - 100 particles per second.

The dead time of the counter is the time during which the counter is unable to detect entering particles.

The discharge may be stopped not only by means of quenching devices, but also by adding alcohols or other organic compounds consisting of complex molecules (methane, isopentane, and others) to the gas filling the counter (usually an inert gas). Usually, the counter is filled with a mixture of 90% argon and 10% alcohol vapor.

Organic molecules strongly absorb ultraviolet radiation produced in the counter. This brings about an abrupt decrease in the number of secondary electrons ejected by this radiation from the cathode. Furthermore, there are loosely bound electrons in these molecules that can be easily pulled off. These electrons neutralize the positive ions by combining with them. The ions of the organic molecules very infrequently eject electrons from the cathode. As a result, the discharge in the counter is stopped. The dead time of the self-quenching counter is on the order of 10^{-4} seconds. It should be noted that as a result of the absorption of ultraviolet radiation, /114
the molecules of the alcohol disintegrate. This gradually leads to the deterioration of the quenching properties of the mixture and to inactivation of the counter.

The Geiger-Mueller counter is used also to detect γ-photons. The detection of photons takes place primarily as a result of the ionization of secondary electrons ejected from the cathode (cylinder of the counter). It is clear that the greater the thickness of the cathode and atomic number of the cathode material, the greater will be the number of secondary electrons ejected from the cathode. Consequently, the efficiency of the counter will be greater, i.e., the ratio of the number of particles detected by the counter to the number of particles incident on the counter. It should be noted, however, that the increase in efficiency of the counter with respect to the γ-photons will be observed only as long as the wall thickness is less than the maximum range of the secondary electrons. A further increase in the wall thickness will tend to decrease the efficiency of the counter, due to the attentuation of the γ-radiation in the wall of the counter. The cylinder of the counter for γ-radiation is usually made either from copper or from copper-plated glass 1 mm thick. The γ-counting efficiency of the counters is small, and is usually 1 - 2%, and in a number of cases — a fraction of one percent.

Scintillation method

It is known that radiation of any type will produce ionization and excitation of atoms and molecules in passing through matter. The transition of an atom or molecule from the excited to a ground state takes place with the emission of photons of visible and ultraviolet radiation. Since in most all solid and liquid substances, there is a strong interaction between individual atoms and molecules, the excitation energy of a single atom is transferred to other atoms, rather than being emitted in the form of light. As a result, the energy of excitation is converted to the energy of motion of the molecules, i.e., heat. However, for some organic and inorganic substances (anthracene, stilbene, zinc sulfide, naphthalene, and others), the probability of light emission in passing to the ground state is greater. This phenomenon is known as luminescence. It is clear that the greater the absorption of the energy of radiation, the greater will be the intensity of light emitted in the process of /115
luminescence. Thus, by recording the number of light scintillations or the total emitted light intensity, it is possible to determine the ionizing particle flux or the exposure dose. Light scintillations are amplified and transformed into electrical pulses by means of photomultipliers. These are then amplified electronically and applied to a detector calibrated in the corresponding units of measurement. The instrument, consisting of the luminescent material and the photomultiplier is known as a luminescent or scintillation counter.

It should be noted that for some materials, the light scintillation has a very short duration, i.e., all the energy of excitation is emitted in the form of light practically at the moment of the passage of radiation through the phosphor (scintillator). This phenomenon is known as fluorescence.

However, there are phosphors that emit only a small fraction of the light at the time of passage of the particle. A large part of the energy spent by the radiation in the excitation of the atoms and

molecules of the phosphor is trapped for a rather long time. This excitation energy may be emitted at any time by the irradiation (stimulation) of the phosphor by infrared or by visible light. This phenomenon, in contrast with fluorescence, is known as phosphorescence. Scintillators capable of storing the excitation energy are known as flash phosphors. Since flash phosphors may emit light only after additional stimulation, they may be conveniently used in dosimetry for the measurement of the total dose of radiation delivered during a given time period.

As a result of the improvement in the technology of the preparation of various phosphors, as well as in the manufacture of highly sensitive photomultipliers, scintillation methods of dosimetry find ever wider practical applications.

Photographic method

Photochemical processes occur in photoemulsions as a result of the action of radiation on photographic film or plate. As a result of these processes, metallic silver is deposited during development /116
in the locations where the absorption of radiation took place. If the photographic emulsion is exposed to highly ionizing particles (α-particles or protons), a track will be formed by the developed grains along the trajectories of the particles. The number of particles can be determined from the number of tracks in the photographic emulsion and the energy — from the length of the track.

The detection of β-particles and γ-photons may be made from the total blackening of photographic film. In order to determine the dose produced by γ-radiation, it is necessary to surround the film by a layer of air or tissue equivalent material whose thickness is equal to or greater than the range of the secondary electrons. The blackening of the photographic film is due to the ionization produced in the photographic emulsion by the secondary electrons resulting from the γ-radiation in the layer surrounding the photographic film. By analogy with the ionization chamber, it is possible to determine

the absorbed energy and, consequently, the absorbed dose in the surrounding layer as well, from the blackening of the film.

It is possible to measure the dose or flux of thermal neutrons surrounded by a layer of cadmium from the blackening due to γ-radiation emitted by thermal capture of neutrons by cadmium nuclei.

The flux of fast neutrons may be determined from the number of tracks produced in the emulsion by recoil protons. In certain cases, various substances are added to the photographic emulsion. The nuclei of these substances capture thermal neutrons and emit charged particles (α-particles, protons) as a result of some nuclear reaction. The flux of neutrons can be determined from the number of tracks formed by charged particles and from the reaction cross section, i.e., the capture probability of neutrons of a given energy by the nuclei of the substance included in the emulsion.

The photographic method is widely used in dosimetry to monitor the dose of γ-radiation received by individual workers.

Chemical method

During the last few years, powerful sources of radiation have /117
been used for experimental purposes, in particular, in the acceleration of chemical reactions, sterilization of bandages and medical compounds, disinfestation of grain, etc. The problem of measuring very large doses of radiation, on the order of thousands and millions of Roentgens, came up in this regard. The previously described methods of detecting radiation are rather sensitive, and are unsuited for the measurement of large doses. Various chemical systems turned out to be the most convenient for these purposes. These systems depend on radiation to produce one change or another; for instance, the coloration of solutions and solids, the precipitation of colloids, and the evolution of gases from compounds. The most widely used method of the several chemical dosimetry methods that have been presently developed is the one based on the oxidation of ferrous to

ferric compounds, the reduction of cerium to the trivalent state, and others. Various glasses which change their coloration under the action of radiation are being developed, and are beginning to be used in the measurement of large doses.

Dosimetric Instrumentation and Dosimetric Monitoring

The safety of work with radioactive substances and sources of radiation can be assured by organizing systematic dosimetric monitoring of internal and external radiation levels of occupational workers, as well as by monitoring the radiation level in the environment. Dosimetric monitoring is one of the principal mainstays of the system of radiation safety. It is clear that the amount of dosimetric monitoring (number and type of dosimeters, as well as the method of distributing them in places of work, and the frequency of measurements of the radiation level) depends on the nature of the work with radioactive substances. If the work is conducted with sealed sources of radiation (as in industrial radiography or in gamma radiation therapy), then it is enough to measure the γ-radiation dose (and neutron dose, if there are neutron sources) at the main and satellite buildings, at places of work that are fully or partially occupied by occupational workers. The monitoring of the radioactive contamination of the air and work surfaces, of the places of work and adjoining buildings, as well as the monitoring of the contamination levels of hands and apparel of workers is necessary in work with unsealed radioactive substances; for example, in radiochemical laboratories, as well as in nuclear reactors, where the leakage of radioactive substances from the system of primary shielding or the appearance of radioactive gases and aerosols is possible. /118

Radiation monitoring instrumentation may be of the fixed, as well as of the portable type. Non-portable detectors are usually placed in buildings where work with radioactive materials is permanently conducted, where radioactive sources (nuclear reactors, irradiation facilities, etc.) are housed, where there are links

between the primary and secondary systems of the nuclear power generating installations, where radioactive wastes are collected and stores, etc.

The instrumentation panel of fixed radiation monitoring devices are located in special buildings. This instrumentation is provided with audible or visible signals which operate when a given radiation level is exceeded.

Fixed radiation monitoring instrumentation is operated continuously, and its primary functional purpose is to signal the disruption of the normal technological processes manifested by the deterioration of the radiation environment. This is evidenced by the increase in either the dose rate of gamma or neutron radiation, or in the concentration of radioactive gases or aerosols in places of work or outdoors.

It is perfectly clear that according to the indications of detectors of fixed radiation monitoring instrumentation regarding the dose rate of γ-radiation, it is not always possible to determine accurately enough the dose received by a worker in a given building, since the level of radiation at various points in the building may vary, especially during maintenance. Therefore, occupational workers directly working with radioactive materials must be furnished /119
individual dosimeters for monitoring γ-radiation doses.

It was mentioned in Chapter 4 that according to NRS-69 personnel dosimeters need not be furnished to individuals (category A) working in a controlled zone under conditions such that radiation loads are known to be less than 1/3 of the MPD. For example, this may apply to individuals working at the control panel of a nuclear reactor or installation, etc. However, fixed radiation monitoring instrumentation must be installed in these premises to signal a change in the radiation environment. Timely preventive measures can then be taken to avoid over exposure of personnel, and to eliminate the causes that led to a deterioration in the radiation environment.

It was also noted that according to the data on the concentration of radioactive gases or aerosols, it is practically impossible to evaluate the intake of radioactive materials into the organism through the respiratory system. Investigation of the radiation environment shows that the concentration of radioactive gases or aerosols in a building varies over a significantly wider range than the dose rate from γ-radiation, and depends on parameters which are difficult to determine, such as intensity of ventilation, configuration and size of buildings, presence of stagnant areas, distribution of gas or aerosol sources relative to the exhaust area, etc. Therefore, in those cases where there is a potential hazard involving the entry of radioactive substances into the organism, one of the prime tasks of radiation monitoring is to determine the content of radioactive substances inside the body by means of special instrumentation of the type SICh (see page 140), or by the activity of excretions (urine, stool). It should be noted that at the present time, individual aerosol samplers have been developed, and are being put to practical use. These samplers measure continuously the concentration of radioactive aerosols in the immediate vicinity of the breathing zone of the worker. Data from individual samplers makes it possible to evaluate more concretely the uptake of radioactive materials by the organism of the worker.

In working with radioactive materials and sources of ionizing radiation, the monitoring of the radiation environment is conducted not only by fixed, but also by portable dosimetric devices. Portable radiation monitoring devices are used when it is necessary to determine the radiation level at the immediate location of work in order to determine the permissible time of occupancy at a given location. The need for determining this arises most frequently in conducting repairs or maintenance, in the collection and transport of radioactive wastes, etc. Furthermore, portable dosimetric devices are used to monitor periodically the radiation environment outside the controlled area. /120

The instrumentation required to conduct radiation monitoring may be divided into the following main types:

— devices for measuring external fluxes of radiation;
— devices for measuring concentrations of radioactive gases and aerosols in air;
— devices for measuring levels of contamination of work surfaces, special apparel, hands, body by radioactive substances;
— devices and methods of personnel dosimetric monitoring;
— devices and methods for determining the content of radioactive materials in the organism.

Examples of several types of radiation monitoring devices finding practical application at present are described in the following sections.

Devices for Measuring γ-Radiation and Neutrons

All devices used to measure the dose of γ-radiaton do not differ much in principle from one another. They consist of a detector made up of an ionization chamber with a preamplifier and an analyzer block made up of a D.C. amplifier, power supply, and an analyzer.

In some cases, devices with gas discharge or scintillation counter detectors are used to measure the level of γ-radiation, as well as the flux of β-particles. These devices are known as radiometers. In spite of the fact that radiometers are significantly less accurate in measuring the dose than dosimeters, they are more sensitive in allowing the measurement of very low radiation doses.

In order to conduct continuous remote monitoring of the radiation environment by means of protable detectors, as well as to monitor technological processes and the technical functioning of our industry, a multi-channel fixed dosimetric instrument "Sistema" [34] is available. It may be modified to provide 50, 100, 150, 200, and 250 channels. This instrument makes it possible to monitor the /121

dose of γ-radiation, fluxes of fast and thermal neutrons, concentrations of β-radioactive gases, and α- and β-active aerosols. The instrument has a device to record the data from the detectors, to signal the exceeding of a given dose, and to maintain verbal communication. Electrical signals from the detectors are input over cables to the central control panel (CCP). Figure 24 shows the control part of the "Sistema" instrument. In order to grasp more clearly the potentials of this system, Table 16 gives the ranges of measurements and thresholds of the detectors designed to measure various parameters of the radiation environment.

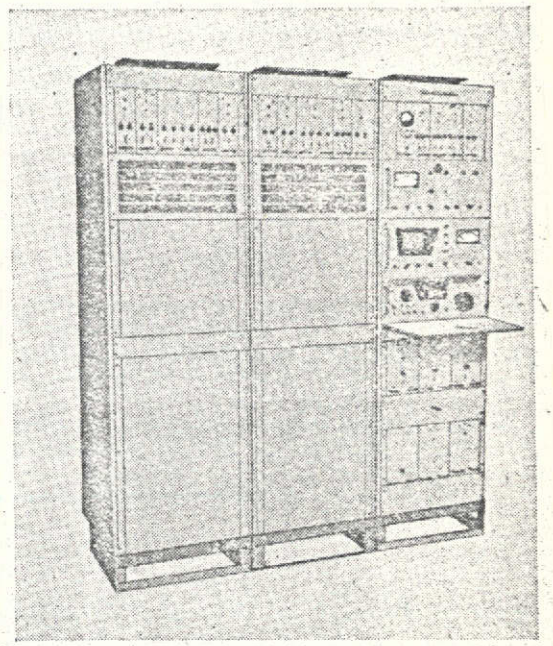

Figure 24. Control panel of the multi-channel dosimetric and technological monitoring device "Sistema" (type 8004-01)

TABLE 16. RANGE OF MEASUREMENTS AND SIGNAL THRESHOLDS OF THE DETECTORS OF THE UNIT""SISTEMA"*

Parameter measured	Measurement range	Signal threshold
γ-radiation dose rate, μR/sec	$5 \cdot 10^{-2}$—$5 \cdot 10^{6}$	$5 \cdot 10^{-2}$—$5 \cdot 10^{4}$
Thermal neutrons flux density, neutrons/(sec · m^2)	10^{4}—10^{8}	10^{4}—10^{6}
Fast neutron flux density, neutrons/ (sec · m^2)	10^{5}—10^{8}	10^{5}—$5 \cdot 10^{6}$
Concentration of β-active gases, Ci/l	$5 \cdot 10^{-10}$—10^{-5}	$5 \cdot 10^{-10}$—$5 \cdot 10^{-7}$
Concentration of β-active aerosols, Ci/l	10^{-13}—10^{-10}	$5 \cdot 10^{-13}$—$2,5 \cdot 10^{-11}$
Concentration of α-active aerosols, Ci/l	10^{-14}—10^{-11}	$5 \cdot 10^{-14}$—$2,5 \cdot 10^{-12}$

*Translator's note. Commas in numbers represent decimal points.

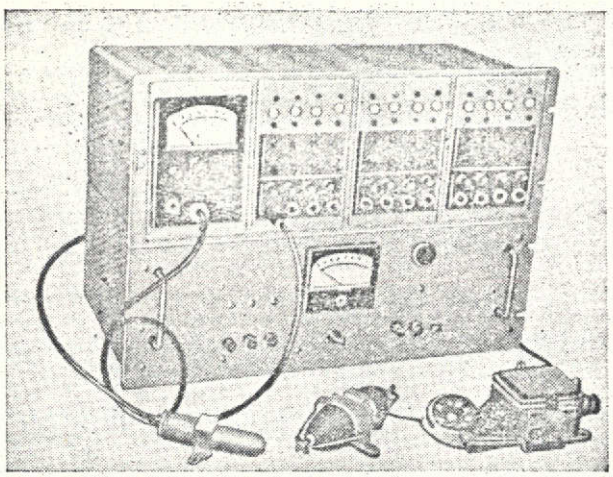

Figure 25. Dosimetric signal analyzer unit USID-12

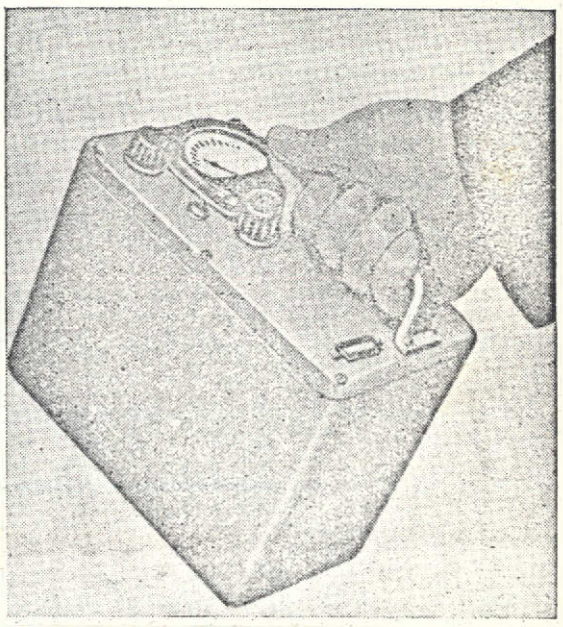

Figure 26. Portable microroentgen meter MRP-1

In order to conduct fixed radiation monitoring on a smaller scale, two or twelve channel devices of the type USID-2, USID-12 [34] may be used. Detectors of devices, such as USID, may be located at distances of up to 200 meters from the control panel. Figure 25 shows our USID-12 device.

A portable type device for the measurement of dose rate of γ-radiation in work places is the microroentgen meter of the MRP-1 type (Figure 26). The detector is a 1-liter ionization chamber mounted on the same chasis with the electronics and meter. It is battery powered; its range is 0.5 - 5000 μR/sec.

A survey type device used to locate radioactive substances is the radiometer SRP-2 (Figure 27). The detector of the radiometer SRP-2 is a scintillation counter. The measurement range is 2 - 1250 μR/hr.

The radiometer RK-01 and its new modification RK-02 (Figure 28) /124 are rather convenient devices to measure the dose rate of β- and γ-radiation at places of work, to verify the effectiveness of the biological shields of reactors and irradiation facilities, as well as to conduct the operational dosimetric monitoring. The sensitivity

of the radiometer RK-02 to the dose rate of γ-radiation is very wide, and ranges from 0.01 to 1000 μR/sec. In contrast with RK-01, the RK-02 radiometer has 5 sub-ranges of sensitivity. Three halogenated counters STS-5 are used as detectors in RK-02 to measure the dose rate

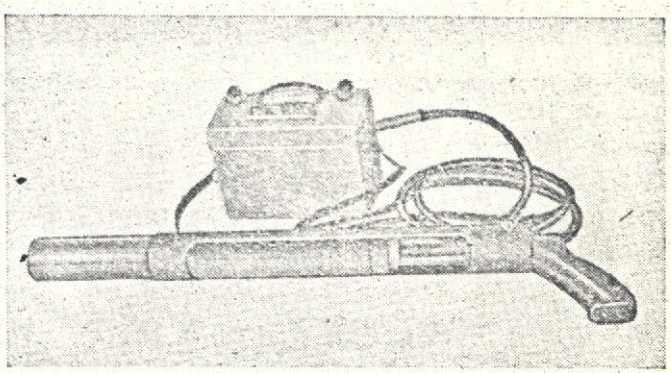

Figure 27. Portable survey radiometer SRP-2

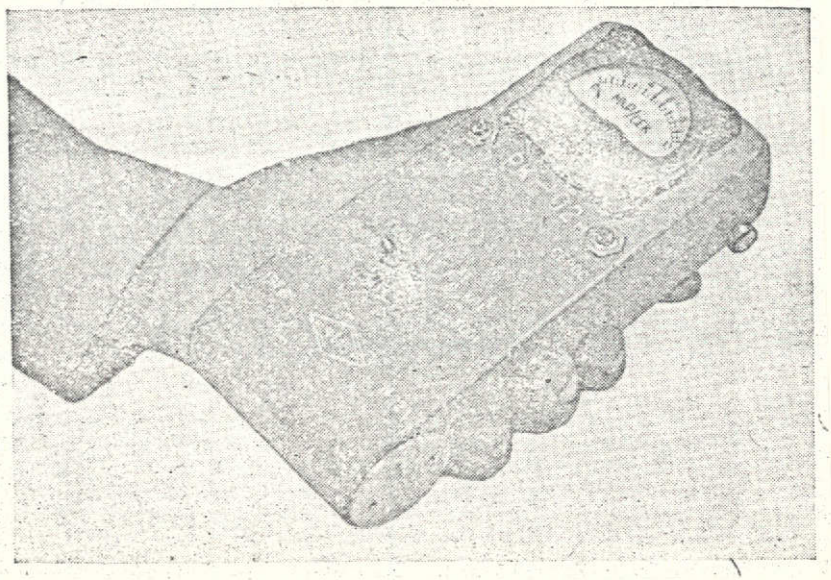

Figure 28. Pocket radiometer, model RK-02

down to 10 μR/sec, and one counter S1-3BG to measure the dose rate of γ-radiation down to 1000 μR/sec.

Fluxes of thermal neutrons may be measured by any of the described dosimeters or radiometers. The detector of the devices must be covered with cadmium covers, and two measurements must be made. The measurement with the jacket determines the dose rate due to γ-radiation and thermal neutrons. Without the cover, the dose rate is due to γ-radiation alone. Once the instrument has been calibrated with thermal neutrons, the dose rate (or flux) of thermal neutrons may be determined from the difference in readings.

The remote monitoring of the flux density of thermal neutrons, intermediate and fast neutrons, in the presence of a γ-background may be made by means of the neutron radiometer of the type "Midiya" (Figure 29). The device consists of an analyzer and five types of detectors. The radiometer includes . . .*

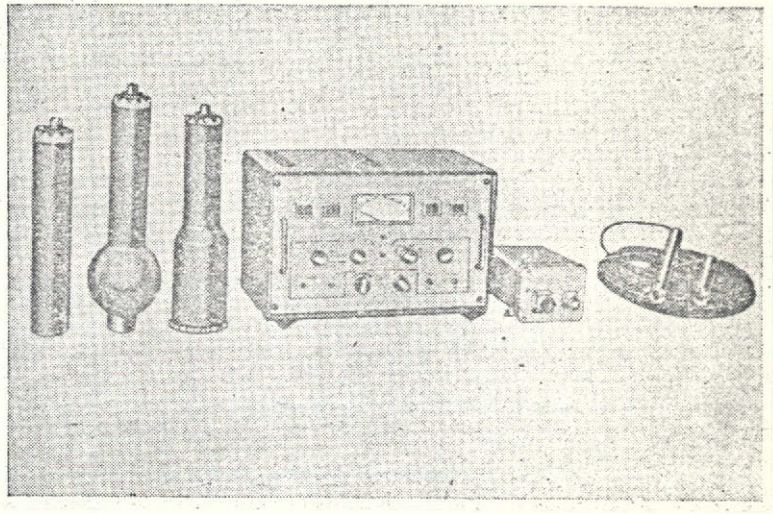

Figure 29. Neutron radiometer "Midiya"

and has an output jack for use with a plotter to record data. The /126 range of measurements of the flux density of thermal neutrons is 10^4 - 10^{10} neutrons/(sec · m^2); intermediate neutrons — (0.03 - 1 MeV) — 10^5 - 10^9 neutrons/(sec · m^2); and fast neutrons — 1 - 14 MeV) — 10^5 - 10^9 neutrons/(sec · m^2).

The dose rate of neutrons, from thermal to 20 MeV energies, may be measured by a neutron dosimeter of the type DN-1A. The device has four sub-ranges which permits the measurement of dose rates in the range from 3 - 3000 mrem/hr.

In conclusion, some mention should be made of the all-purpose radiometer, model RUS, for the measurement of the neutron dose of any spectral composition, including intermediate neutrons. This device has detectors capable of measuring the dose rate of γ-radiation, fluxes of fast and thermal neutrons, the level of contamination of surfaces by α- and β-active substances in the presence of γ-background of up to 40 R/sec (Figure 30) [36].

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*Translator's note. Apparent omission in the original foreign text at transition from page 125 to 126.

Furthermore, the device Model RUS has a detector which is also capable of determining the Sr^{90} activity of teeth. As a result, the content of this isotope in the organism can be evaluated. The device is battery powered; the mass of the device without the counters is 5 kg.

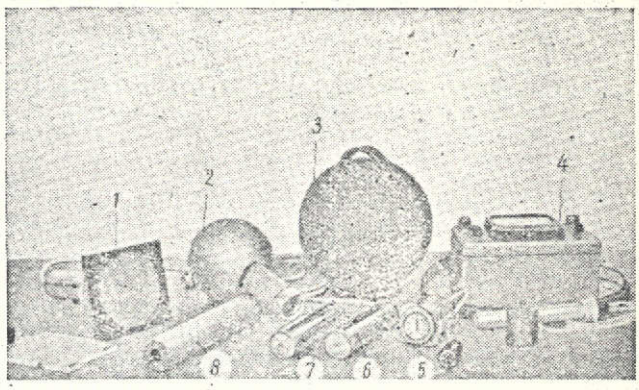


Figure 30. All-purpose scintillation radiometer RUS-7:

1 — detector for measuring contamination of surfaces by β-active materials; 2 — moderator covering the radiometer 6 in measuring the flux of intermediate and fast neutrons; 3 — moderator for use with radiometer 6 in measuring the dose rate of mixed spectrum neutrons; 4 — measuring panel of the instrument (a gas discharge counter SI-10BG is mounted on the panel to measure the dose rate due to γ- radiation); 5 — electromechanical scaler; 6 — radiometer; 7 — detector for measurement of thermal neutrons; 8 — cadmium jacket

Measurement of Atmospheric Contamination by Radioactive Aerosols

Since mean annual permissible concentrations of atmospheric radioactive aerosol contaminants in the work buildings are very small, it is not possible to determine directly the activity of aerosols in air by any given counter. Therefore, it is necessary to concentrate aerosols in air on small areas of various filters. Knowing the volume of the filtered air and having determined the activity of aerosols deposited on the filter, it is possible to determine the concentration of aerosols in air. Paper filters, cotton, special benzene filters, and others may be used. One of the best filters is a special tissue, Model FP, which almost completely traps all aerosols, i.e., it has a low transmission coefficient and is not very thick. The optimum flow rate for these filters is 1 $l/(min \cdot cm^2)$. This flow rate give a minimum transmission coefficient.

Any set of instruments for the determination of the concentration of radioactive aerosols consists of the following basic

components: detector with filter, device for measuring the flow rate of air, air pump, and counters for determining the absolute activity of aerosols deposited on the filter.

Fixed devices for continuous monitoring of the conentrations of α- and β-radioactive aerosols in the air are the aerosol radiometer models ARS-2 and RA12S-1 (Figure 31) [34, 35]. In these devices,

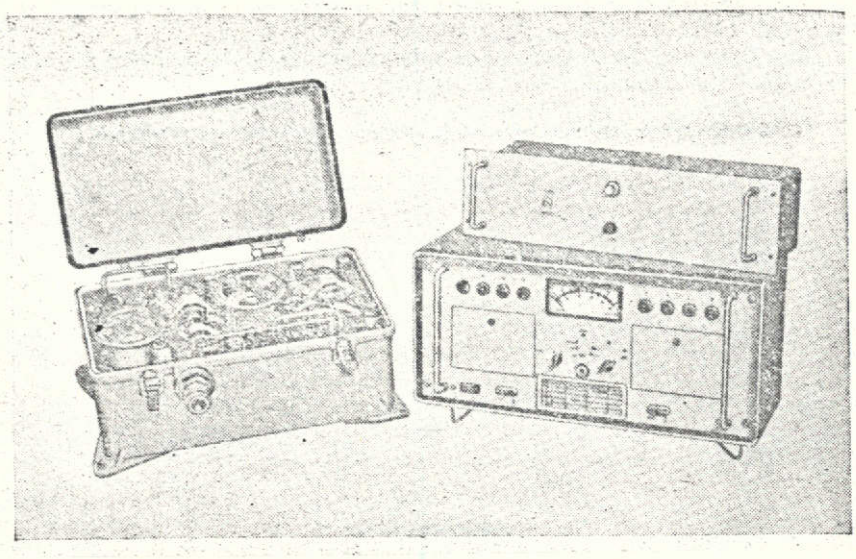


Figure 31. Aerosol radiometer RA12S-1

the aerosols are sucked in through a continuously moving filter ribbon by the air collector. The activity of the aerosols deposited on the ribbon is recorded by detectors of α- and β-radiation. Pulses from the detectors are channeled to the signal analyzer. The instrument has a plotter and a device that provides a signal whenever a given concentration of radioactive aerosols is exceeded. An automatic γ-background subtraction of up to 1 mR/hr is provided. The filter ribbon may be moved with a velocity from 0.1 to 10 mm/min. At a ribbon feed rate of 0.5 mm/min, and an air flow rate of 100 ℓ/min, the range of measurements of the α-active aerosols is between 10^{-14} - 10^{-10} Ci/ℓ, and β-active aerosols — 10^{-13} - 10^{-10} Ci/ℓ. /129

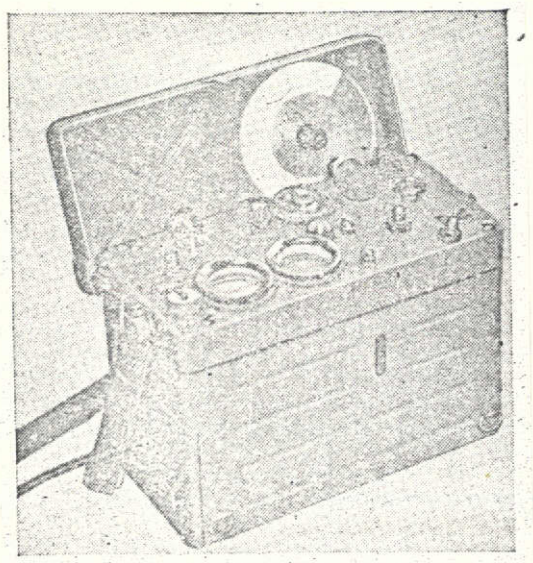


Figure 32. Portable RV-4 unit for the measurement of the radioactive aerosol concentration in air

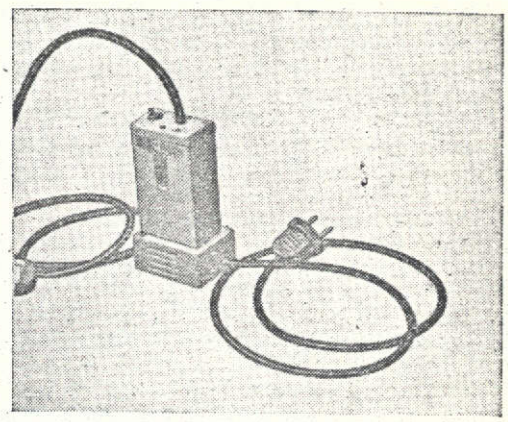


Figure 33. Personnel air sampler for measurement of aerosol concentration

Instrument Model RV-4 (Figure 32) is used as a portable unit to measure the concentration of α- and β-active aerosols. It is powered by connection to an A.C. line.

The personnel sampler (Figure 33) is used to measure the concentration of radioactive aerosols in the immediate breathing zone of the worker. Standard filters are used to collect the samples. The unit consists of a filter holder in the shape of a light aluminum box, and a membrane pump.

Devices for the Measurement of Surface Contamination

The contamination of surfaces by radioactive materials is measured by the number of α- or β-particles emitted from a given area of the contaminated surface. This is convenient, since correction factors, which are difficult to determine, are not required for backscatter and selfabsorption of α- or β-particles in the active layer of the contaminated surface. At the same time, the calibration of instruments is simplified, and a rapid determination of the surface contamination can be made. /130

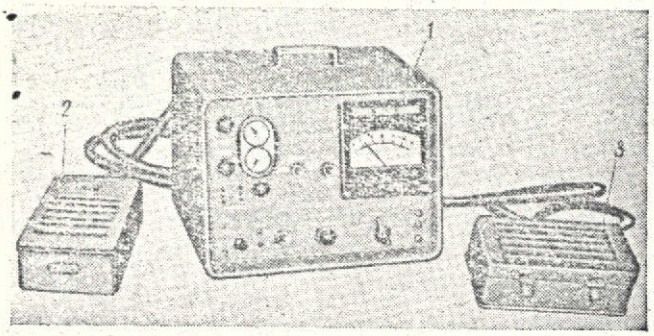


Figure 34. All-purpose radiometer model TISS for measurement of contamination of surfaces and clothes:

1 — measuring panel; 2 — detector TG for measurement of contamination by β − γ-active substances; 3 — detector TYu for measurement of contamination by α- active substances

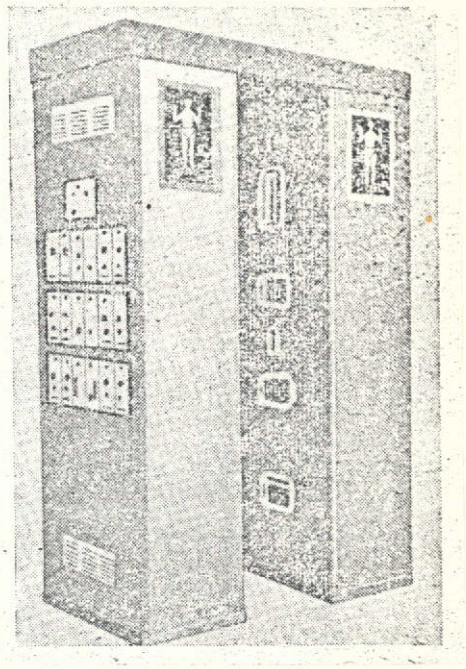


Figure 35. Su-1 unit for measurement of contamination level of radioactive substances on special apparel and body

Detectors of instruments designed for the measurement of surface contamination are usually gas discharge or scintillation counters. The most widely used instrument is the all-purpose radiometer model TISS, with 3 detectors. The detector model TG, with 3 counters STS-6, is designed for the measurement of surface contamination by β- and γ-active materials; the detector TYu is an air filled proportional counter for the measurement of contamination of surfaces by α-active materials (Figure 34). The measured area of these detectors is 150 cm^2. The third detector of the instrument TISS, TI, is a scintillation counter with a photomultiplier to measure the contamination of small surfaces by α-active materials; the sensitive area is 23 cm^2.

To measure the contamination level of special apparel and body, the Su-1 unit is currently used. This unit automatically signals /131 whenever the level of contamination of the body and apparel by β- active materials is surpassed. The detectors of the unit SU-1 are gas discharge counters STS-6 (Figure 35) [35]. The unit SU-1 is usually placed at the exit from the controlled zone and from the sanitary passage.

Methods of Individual Dosimetric Monitoring

In working with radioactive materials, it is important to know the dose of radiation received by the worker during a given time period. On the basis of these data, one may determine the action of radiation on each individual worker. Personnel or, as they are /132
known, pocket dosimeters are used to conduct personnel dosimetric monitoring. These are attached to the special apparel.

Personnel dosimeters using special types of photographic film (method of IFK and IFKU), small ionization chambers (method IDK), and thermoluminescent crystals (method IKS) are currently used to monitor γ-radiation [37 - 39].

The photographic dosimetric monitoring is based on the comparison of optical densities of the darkening of the film with control films which were exposed to known doses of γ-radiation. The photographic film is placed in a special light-proof plastic cassette. In order to eliminate the dependence of the film darkening on the energy of radiation (on hardness), the cassette is surrounded by a lead filter 0.75 mm thick. The degree of darkening of individual and control films, after development and fixing, is determined by a densitometer.

The construction of the cassette is modified in IFKU [37]; this modification makes it possible to record on photographic film not only the dose delivered by γ-photons, but by β-particles and thermal neutrons as well. Figure 36 shows the external appearance of the IFKU cassette, which consists of a body, filters pressed into it, /133
lock covers, and pins for fastening the cassette to the clothes.

Depending upon the specific work, the cassette is given out for one of two weeks, and sometimes even for a month, i.e., this method is capable of determining the integral dose of radiation received by the workers while wearing the cassette.

The photographic method of personnel monitoring is capable of determining radiation doses between 0.05 to 2 rem, by using RM-1, "Roentgen XX", and "Agfa" film. The upper limit of measuring dose may be increased to 15 - 50 rem by using RM-5-3 and RM-5-4.

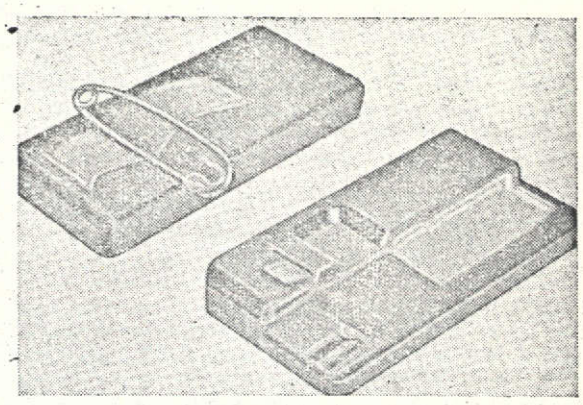


Figure 36. External appearance of the IFKU cassette

Personnel dosimetric monitoring is also carried out by means of small ionization capacitor chambers. Before being handed out, the chamber is charged to a given potential. Ionization is produced in the chamber by radiation, and as a result the potential is decreased. The dose received by the worker is determined from the difference between initial potential and the resulting potential after exposure in an area subjected to radiation. The charging of the chamber and the determination of the potential after exposure is made in a special measuring unit. Figures 37 and 38 show the general appearance of the charge-measuring unit and the capacitor chambers, which are included in the KID-1. These chambers are capable of measuring γ-radiation in the range from 0.02 to 2 R.

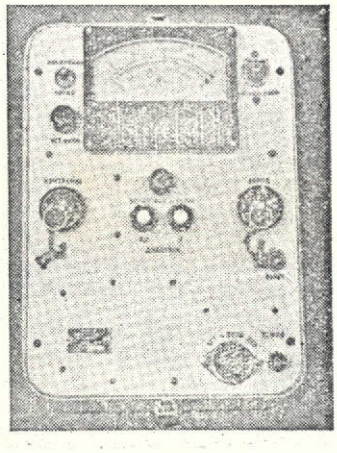


Figure 37. Charge-measuring unit KID-1

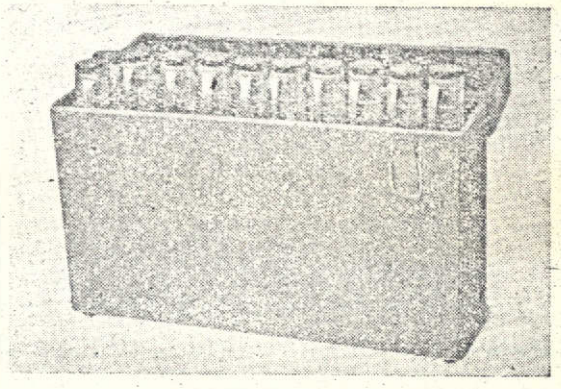


Figure 38. Capacitor chambers of the KID-1 unit

In addition to the KID-1 personnel dosimetric monitoring instrument, other ionization chambers KD-1, ID-1, DK-02, and others are used. They differ in the range of measured doses and in a few structural details of the chambers and measuring panel.

The method of personnel monitoring IKS, developed recently, /134
appears to be rather promising [38, 39].

The detector of radiation is a special type of so-called thermoluminescent crystals. Under the action of radiation, the molecules of the crystal pass to an excited state. This excitation energy is stored in the crystal dosimeter, and is extracted in the form of a light pulse after heating the dosimeter to a temperature of 300°. The received radiation dose may be determined from the intensity of the light pulse recorded by special instrumentation.

The advantage of this method, in contrast with other methods of personnel dosimetric control, is the large range of measured doses (from 0.1 to 1000 rad). This property makes it possible to use the method in carrying out repair operation, as well as in determining the dose delivered over a large period of time (months, years) as, for example, during the cruise of the atomic icebreaker "Lenin". Furthermore, the advantage of the method consists in that the dosimeter stores almost forever the absorbed energy of excitation. The readings of the dosimeters are practically independent of the temperature and humidity at which they were exposed.

Monitoring of Internal Exposure

In nuclear reactors or in radiochemical laboratories, the poten- /135
tial hazard exists that radioactive materials may penetrate into the organism as a result of violations of the safety rules. Therefore, along with doses produced by external fluxes of radiation, which may be determined by personnel dosimeters, it is important to know the dose of internal exposure, which may be calculated if the composition and quality of the radioactive materials located in the organism are known.

The content of radioactive materials in the organism may be determined by the measurement of the activity in excretions (urine, stool). It is known that the permissible amount of radioactive materials in the organism is very small, and varies for different isotopes in the range from tenths to tens of microcuries. It is clear that the excretions contain a small portion of the total activity in the whole body, a quantity which is rather difficult to measure. Therefore, the radiometric measurement of excretions is only made in accident situations, when the possibility exists of the entry into the organism of a significant amount of radioactive materials. It should be noted that this method, in spite of its low sensitivity, is the only available one for the determination of the α-active isotope content in the organism.

At present, several highly sensitive instruments of the type SICh (counter of radiation from humans) have been developed to determine the content of β- and γ-active isotopes in the organism. Large areas(140 - 200 mm diameter) scintillation crystals or liquid organic scintillators, surrounded by massive shielding to reduce natural background and, therefore, to increase the sensitivity of the instrument, are used to detect radiation.

The person is in a chamber between detectors, which are brought as close as possible to the body to increase the sensitivity. The identity of the isotopes in the body is determined from the spectrum, while the activity is determined from the intensity of radiation. The activity of γ-active emitting isotopes (Cs^{137}, Co^{60}, K^{40}) is determined directly, the activity of β-emitting isotopes (P^{32}, Sr^{90}) — from their bremsstrahlen radiation. Instruments of the SICh type may be used to determine hundredths parts of the permissible amount of radioactive substances inside the organism [40].

CHAPTER 6

ORGANIZATION OF OPERATIONS WITH RADIOACTIVE SUBSTANCES AND RADIATION PROTECTION

In working with radioactive substances and sources of ionizing radiation, the proper organization of operations providing for the safety of the staff, individuals from the population, and the population as a whole is of primary importance. The proper organization of operations with radioactive substances means specifying the set of measures for ensuring radiation safety, for which the radiation load from sources of external and internal radiation does not exceed the regulation maximum permissible dose (MPD) or the limits of doses for the corresponding categories of individuals and groups of critical organs. These measures include the creation of protection from external radiation flux, the prevention of the propagation of radioactive substances into working areas and into the environment, the proper planning and isolation of accommodations, the organization of the required radiation monitoring and healthful conditions, the provision of the required conditions for transporting radioactive substances, the collection and burial of radioactive waste, the use of individual shielding methods, the performance of decontamination operations, etc. /136

The guiding document regulating the requirements for providing radiation safety in working with radioactive substances and sources of ionizing radiation is "Basic Health Rules for Working with Radioactive Substances and Other Sources of Ionizing Radiation", PSR-72 No. 950-27 [28].

The creation of PSR-72, in place of "Health Rules SR-333-60" [24] previously in effect, resulted from the necessity of correcting a number of provisions in connection with the implementation of NRS-69. Moreover, on the basis of accumulated experience in working with radioactive substances and sources of ionizing radiation, it

appeared necessary to review and refine a number of provisions of the rules previously in effect. In particular, the values of the regulated dose intensity behind shielding for accommodations for various purposes are reconsidered in PSR-72, and also the allowable levels of contamination of individual shielding methods; a number of /137 provisions are made more precise, which touch on the arrangement of laboratories, enterprises, and institutions where operations are carried out with radioactive substances or sources of ionizing radiation, the processing and burial or radioactive waste, the organization of radiation monitoring, the requirements on decontamination of working areas and equipment, etc.

The most general provisions providing for radiation safety, which are appropriate and compulsory for all enterprises and institutions where operations are carried out with radioactive substances and sources of ionizing radiation are discussed in PSR-72. Thus, they are called "Basic Health Rules".

It is quite obvious that development, addition or change in the individual provisions providing for radiation safety can be required, depending on the specifics of the operation. Hence, it was indicated in the introduction to PSR-72 that, on the basis of the present "Rules" and NRS-69, the Ministries and Departments must develop new rules or introduce changes in the existing rules according to the individual problems of performing operations with the use of radioactive substances and other sources of ionizing radiation [28].

The basic requirements on providing radiation safety are briefly discussed in this chapter, and an interpretation is given on the most important, from out point of view, provisions of PSR-72.

Protection from External Radiation Flux

For any operations with radioactive substances, man is subjected to the effect of ionizing radiation. Consequently, the problem consists of creating protective enclosures which would decrease the dose of external irradiation to the maximum permissible level.

In view of the fact that the range of α-particles emitted by radioactive substances is very small, it is not necessary to indicate protection from external irradiation by α-particles, even if operations are carried out with exposed radioactive substances. It is sufficient to be located at a distance of 9 - 10 cm from the radioactive preparation, and no α-particles will strike the worker, since the range in air of α-particles emitted by radioactive substances does not exceed 8 - 9 cm. Clothes and rubber gloves are complete protection from external irradiation by α-particles. /138

In order to protect the worker from external irradiation by β-particles, it is necessary to carry out operations with radioactive substances behind special screens or in special shielded cabinets. The thickness of the protective screens must be greater than the maximum range of the β-particles. β-active substances should be contained in vessels or containers with the appropriate wall thickness. Plexiglass, aluminum, or glass are used as shielding materials.

The shielding thickness can be determined from the following approximate formula:

$$d = (0.54 \cdot E_{max} - 0.15) \text{ gm/cm}^2, \qquad (53)$$

where E_{max} is the maximum energy of the β-spectrum of the given radioactive isotope, MeV.

The situation is more complex with the creation of protection from external flux of γ-radiation and neutrons, whose penetration is significantly greater than α- or β-particles. Moreover, as was indicated in Chapter 2, in contrast to α- and β-particles, it is impossible to completely absorb γ-radiation and neutrons, and they can only be attenuated by a specified number of times.

Shielding from γ-radiation and neutrons is rather cumbersome, and its thickness can reach several tens and even hundreds of centimeters — for example, nuclear reactor shielding. Thus, it is recommended shielding design take into account the purpose of the adjacent

locations, the occupation time by workers, and also the category to which the irradiated personnel belong. The shielding must then be designed with a safety factor of 2. Thus, the dose rate behind shielding P must equal:

$$P = \frac{D}{2t} \text{ mrem/hr}, \qquad (54)$$

where D is the dose (mrem/week) corresponding to the MPD for the given category of personnel [28]. Let us consider, for example, the maximum dose rate behind shielding in a location where workers belonging to category A are found no more than t = 18 hours per week, i.e., for which the MPD equals 5 rem per year, or accordingly, D = 100 100 mrem/week. It follows from Expression (54) that:

$$P = \frac{100}{2 \cdot 18} = 2{,}8 \text{ mrem/hr}.$$

TABLE 17. DESIGN DOSE RATE P BEHIND SHIELDING IN AREAS OCCUPIED BY PERSONNEL (CATEGORY A)

Purpose of area	P, mrem/hr
Areas of continuous occupation (t = 36 hr/week)	1.4
Areas in which the occupation time by personnel $t \leq 18$ hr/week	2.8
Unattended areas	28
Any other areas of a given institution	0.1

Table 17 presents the design dose rates P, regulated by PSR-72, /139 behind shielding for areas continuously occupied by personnel (category A) for a 36-hour week, areas where personnel are located no more than half the working day ($t \leq 18$ hr/week), and also for unattended areas and any other areas of a given enterprise or institution. Unattended areas include pits, isolation cubicles, and other areas in which there are sources of radiation — for example, the tubing system of a reactor, etc. — where personnel go sporadically for performance of repair operations only.

As already indicated, for other times t of personnel occupation in areas, the design dose rate behind shielding is calculated from Formula [54].

It follows from Table 17 that the regulation dose rate behind shielding in unattended locations is 20 times higher than in areas of continuous occupation by personnel. This implies that if over one year, the time of operation in these locations does not exceed 1.8 hr/week (or $\sim$ 94 hr/yr), then the annual radiation load (without consideration of the safety factor) does not exceed 5 rem. Situations can arise when operations are required to be carried out in unattended locations for a time greater than that for which the radiation load equals the MPD. In this case, measures should be taken for supplementary screening of the radiation sources, or the operation time should be regulated in an appropriate manner.

According to NRS-69, in organizing operations with radioactive substances and sources of ionizing radiation, and in designing shielding for personnel working at a given enterprise or institution in the vicinity of areas where work with radiation is carried out, one should start from an irradiation dose of 0.5 rem per year, i.e., one should take D = 10 mrem/week in Formula (54). Consequently, in any locations of a given enterprise or institution where work is not carried out directly with radioactive substances, the dose rate behind shielding must be 10 times less than in areas of continuous occupation by personnel, and equals 0.14 mrem/hr, or (rounding to the first significant digit), 0.1 mrem/hr (cf. Table 17). /140

In locations and on the grounds within the zone under supervision, the design dose rate is regulated to be 0.03 mrem/hr, i.e., about 40 times less than for locations of continuous occupation by personnel [28]. This level is established starting from the fact that the MPD for individuals from the population is 10 times less (0.5 rem per year) than for personnel, and an occupation time t = 24 hr/day, i.e., about 4 times greater than in locations of continuous occupation by personnel.

In designing shielding of areas where, for example, gamma-therapeutic or defectoscopic installations are located, one should take into account the possible direction of emission of the radiation beam, and design protection from direct radiation only in these directions; in the other directions it is sufficient to provide protection only from scattered radiation.

Along with the requirements imposed on the dose rate in adjacent locations, it is indicated in PSR-72 that for technological monitoring devices, which must often be used under operating conditions in public areas, the dose rate at a distance of 1 meter from the device unit with the radioactive source must not exceed 0.3 mrem/hr. If one is located at this distance for the entire working day (t = 6 hr/day of 36 hr/week) for the entire year, i.e., 52 weeks, then the annual irradiation dose with equal $D = Pt = 0{,}3 \cdot 36 \cdot 52 \approx 500$ mrem, i.e., the MPD for individuals from the population.

Thus, in using equipment and apparatus at a distance of 1 meter from which the dose rate of 0.3 mrem/hr is produced in working and non-working conditions, there are no special requirements on the areas and position of the equipment. /141

The allowable dose rate close to such equipment is 10 mrem/hr, according to PSR-72.

If the dose rate from irradiating equipment exceeds 0.3 mrem/hr at a distance of 1 meter from the protective case, such equipment must be located in special areas, preferably in separate structures or a separate wing of the structure. There are then no special requirements imposed on the finishing of the areas.

In order to provide access to the area where the irradiating equipment (gamma-therapeutic or defectoscopic) are located for placement of the source in the storage condition (the equipment is shut off) and to carry out the necessary adjustment or repair operations, the regulated dose rate at a distance of 1 meter from the protective

case must not exceed 3 mrem/hr, which corresponds to a MPD for personnel of 5 rem per year.

The control panel of irradiating equipment must be located in an adjacent area. The entrance door to the area where the equipment is located must be provided with a blocking device which prevents entry to the area when the equipment is operating. Moreover, a device must be provided for the forced remote transfer of the source to the storage condition in the event of any emergency [28].

In calculating the shielding devices, it is necessary to take into account primarily the spectral composition of the radiation, its intensity, and also the distance from the source at which the service personnel are located, and the occupation time within the sphere of influence of the radiation.

One can calculate the shielding thickness by using the attenuation law for a wide beam of γ-radiation [cf. Formula (27)]

There are well known at present attentuation tables based on available calculation and experimental data, and also various types of nomograms which permit determining shielding thickness for γ-radiation of various energies [6 - 11].

In principle, one can use any substance as material for shielding from γ-radiation. However, in selecting a shielding material, /142
it is necessary to supervise its structural properties and also the requirements on the dimensions and mass. For the protective case of various types of equipment (gamma-therapeutic or gamma-defectoscopic), where the mass plays the essential role, the most favorable shielding materials are those substances which best attenuate γ-radiation. As was shown in Chapter 2, the greater the density and atomic number of the substance, the higher the degree of attenuation of γ-radiation. Thus, lead is most often used for these purposes, and sometimes even uranium. In this case, the shielding thickness is less than with the use of another material and, consequently, the mass of the protective case is also less.

In creating stationary shielding (i.e., shielding of the area where operations with gamma-sources are carried out) which provide safe occupation by persons in adjacent rooms, it is most economical and suitable to use concrete. If we are dealing with long-wave radiation, where the photoeffect plays the essential role, substances with large Z, in particular, barite, are added to the concrete, which permits decreasing the shielding thickness.

High-power equipment for radiation processing of various substances (sterilization, insect extermination, radiochemical processes, etc.) have found wide application in recent years. They ususally consist of two basic units: a chamber where the irradiation is carried out, and a storage vault where the preparations are placed after the irradiation cycle. Concrete is usually used for the chamber shielding. Water is often used as the shielding material for the storage vault, i.e., the preparations are immersed in a basin with water, whose thickness provides the required decrease of the radiation dose to safe levels. It is more convenient to carry out loading and reloading of the equipment, and also to perform repair operations with water shielding.

In some cases, the conditions for working with sources of γ-radiation can be such that it is impossible to create stationary shielding (for reloading the equipment, recovery of the radioactive preparation from the container, calibration of the device, gamma-defectoscopy with an exposed source, etc.). Bear in mind here that the source activity is small. In order that the personnel be safe from irradiation, one must make use of, as is said, "protection by distance" and "protection by time". This means that all manipulations with exposed sources of γ-radiation should be carried out with /143
the help of long tongs or holders. It is known that for sources of small linear dimensions, the radiation dose decreases inversely proportionally with the square of the distance. Moreover, any operation must be carried out only for that interval of time during which the dose received by the worker does not exceed the established value (cf. Chapter 4). It is necessary to carry out such operations under the control of a radiation supervisor. Then extraneous persons

must not be found in the area, and it is necessary to enclose the zone where the dose exceeds the maximum allowable over the operating time.

If the source activity is less than 0.1 mg-equiv Ra, it is not necessary to provide any shielding measures. One can operate with sources of activity up to 10 mg-equiv Ra in ordinary locations not having special protective walls and floors; shielding is provided (in the form of screens or "protection by time" and "protection by distance") only for persons directly working with the radiation sources.

It should be noted that it is necessary to carry out periodic monitoring of shielding with the help of dosimetric devices, since it can partially lose its shielding qualities with time because of the appearance of some insignificant failure of its integrity, for example, cracks in concrete and barite-concrete enclosures, dents and cracks in lead sheets, etc.

Calculation of neutron shielding is carried out from corresponding formulas or nomograms. Substances with small atomic number should be taken as the shielding materials in this case, since with each collision with a nucleus, the neutron loses the larger part of its energy, the closer the mass of the nucleus to the mass of the neutron (cf. Chapter 2). Water and polyethylene are ususally used for neutron shielding. By losing its energy in the shielding during the process of interaction with atomic nuclei, the fast neutron is converted into a thermal neutron, which is captured by the atomic nuclei, which then emits a γ-quantum. There are no pure neutron beams in practice. As is well known, neutron sources are nuclear reactors, accelerators, and radium-beryllium preparations. Besides neutrons in all these sources, there is intense γ-radiation, which is formed in the reactor during the fission process, and also because of the decay of fission products. In accelerators and radium-beryllium preparations, γ-radiation occurs with various nuclear reactions. Moreover, the decay products of uranium emit γ-quanta. Thus, in /144

designing neutron shielding, one must always simultaneously provide γ-radiation shielding.

Organization of Operations with Radioactive Substances in Exposed Form

In the preceding section, we considered problems of shielding for operations with closed sources of radiation, i.e., when man can be subjected only to external irradiation. Here, the main requirement on providing safe working conditions is the installation of protective enclosures which provide a decrease of the dose of external radiation flux at the working areas and in adjacent locations to the maximum allowable levels, or the use of "protection by time" or "protection by distance", in order that an over-irradiation of the service personnel does not occur during performance of some operation.

Radioactive substances are also used in exposed form — for example, in radiation-chemical operations, extraction of some isotope from a mixture of radioactive products, preparation of radioactive solutions, the use of isotopes in the tracer method, experimental works related to inoculation of animals with radioactive substances, etc. In these cases, the incidence of radioactive substances within the organism is possible, in addition to the external irradiation. Thus, in operations with radioactive substances in exposed form, along with the organization of external irradiation shielding, one should provide a set of measures preventing radioactive contamination of the air and surfaces of the working and adjacent areas, clothing and skin, and also objects of the environment. This set of measures must certainly also be provided at nuclear reactors and radiochemical factories, where there is potential danger (in particular, in performing repair operations) of the penetration of radioactive substances into the working areas and into the environment. These requirements are discussed in detail in PSR-72.

One of the important measures in providing radiation safety when working with radioactive substances in exposed form is the appropriate planning and finishing of locations, for which the /145

propagation of radioactive substances over other areas is made as difficult as possible. Operations with radioactive substances should then be concentrated in one part of the structure, and passage through these areas reduced to a minimum. The finishing and equipment of the areas must ensure their simple decontamination. Moreover, PSR-72 imposes a number of requirements on the ventilation system, organization of the working places and technological conditions, on the system for collection and disposal of radioactive waste, observance of personal hygiene, etc., which permits eliminating the possibility of the incidence of radioactive substances within the organism, and thereby provides safe working conditions. These requirements depend on the nature of the work being carried out with some radioactive isotopes, their activity, and also the group of radiotoxicity to which these isotopes belong (cf. page 107).

Operations with radioactive isotopes are divided into three classes, depending on the radiotoxicity of the isotope and its activity at the working place. For each class of operations, there are corresponding requirements imposed on the finishing of the locations and on other health measures which must be observed. Table 18 presents the activities at the working place for the various classes of operations, depending on the radiotoxicity group to which the given isotope belongs.

TABLE 18. ACTIVITY AT WORKING AREAS FOR VARIOUS CLASSES OF OPERATIONS, μCi [28]

Radiotoxicity group	Class of operations		
	I	II	III
A	More than 10^4	From 10 to 10^4	From 0.1 to 10
B	More than 10^5	From 10^2 to 10^5	From 1.0 to 10^2
C	More than 10^6	From 10^3 to 10^6	From 10 to 10^3
D	More than 10^7	From 10^4 to 10^7	From 10^2 to 10^4
E	More than 10^8	From 10^5 to 10^8	From 10^3 to 10^5

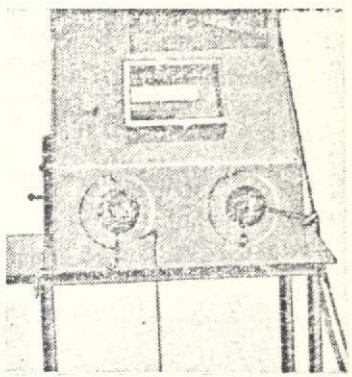
When it is possible to select the radioactive substances for operations, the substance of least radiotoxicity should be used. Moreover, it is necessary to limit the operation such that the amount of the substance at the working place is the minimum necessary for the given operation. It is also recommended to reduce as much as possible the number of such operations as interspersion of powders, sublimation, etc., which are related to losses of the radioactive substance. /146

Class III operations can be carried out at the same locations in which operations with stable isotopes are carried out. The finish of the locations must be the same as in ordinary chemical laboratories (the walls are painted to a height of 2 meters with oil point). Apparatus for washing and isolating locations for storage and packing of radioactive substances and solutions is recommended. Operations with radioactive substances should be carried out on separate tables or even in separate locations. To provide easier cleaning of an area from possible radioactive contaminations, it is desirable (but not compulsory) to cover the floor, and also the laboratory tables, with linoleum, plastic, or other nonabsorbing materials. Operation with volatile, gaseous, or powdered radioactive substances is suitably carried out in exhaust hoods of the ordinary type. In order to eliminate contamination in the hood, it is best to have special trays which are easily washed. Service personnel must be provided with coveralls and caps of bleached moleskin, and also rubber or vinyl chloride gloves.

Specially equipped isolated areas must be provided for Class II operations. The floor should be covered with plastic, and the walls with oil paint to the top. Operations with radioactive substances must be carried out in shielded chambers (isolation cubicles) or special exhaust hoods. Shielded chambers of various types have been manufactured by domestic industry. They provide protection of personnel from external β- and γ-radiation, and also from contamination of the air of the working areas by radioactive gases and aerosols. Rubber gloves or various types of tongs, which permit carrying out

the necessary operations with radioactive substances, are mounted in the front wall of such shielded chambers. The outward appearance of one of the shielded chambers is shown in Figure 39.

Shielded chambers and exhaust hoods are equipped with ventilation. The air moves along an independent system of air ducts, and the removed contaminated air is cleaned at an individual filter at the chamber or in the exhaust hood. This prevents contamination of the system of exhaust ducts and ventilators. Underpressure must be produced in the chambers and hoods to eliminate leakage of air into the laboratory areas. It is necessary for class II operations to carry out systematic dosimetric monitoring of the γ-radiation level in the laboratory, individual monitoring, and also monitoring of the contamination of the air and work surfaces.



/147

Figure 39. Shielded chamber with tong manipulators mounted in the front wall of the chamber with the help of ball joints

As part of the areas where class II operations are carried out, it is necessary to have a sanitary passage or shower, at the exit of which monitoring of the contamination level of the hands and body is carried out.

For class II operations, personnel must be provided with coveralls, caps, gloves, light boots and, when necessary, means for protecting the respiratory system. When performing operations with radioactive solutions or powders, when the probability of contamination increases, or when cleaning the areas, personnel must be provided additionally with plastic aprons and oversleeves or plastic coveralls, and also additional rubber or plastic overshoes.

Class I operations are carried out in areas located in a separate part of the building, and isolated from other areas, and having special planning which provides isolation of the apparatus from the

equipment which are sources of contamination of the air and surfaces, from continuously occupied areas.

It is most rational to use a triple-zone planning.

Chambers, cubicles, unattended areas, where the technical equipment and communications are located, which are the main sources of radioactive contamination, are located in zone I. /148

Periodically serviced areas, where equipment repair and other operations related to opening technical equipment can be carried out, units for loading and unloading radioactive materials, and areas for temporary storage of radioactive waste belong to zone II.

Areas intended for continuous occupation by personnel — operational control panels, etc. — belong to zone III.

A sanitary sluice is used between zones II and III in order to eliminate the possibility of distributing radioactive contamination from zone II to zone III. There must be a sanitary passage of the compulsory type at the exit from zone III.

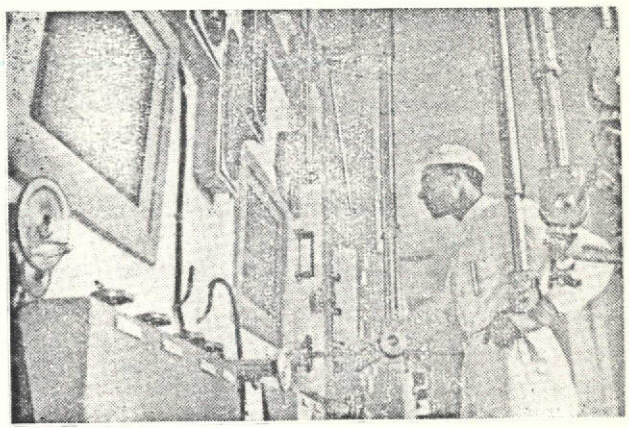
In areas where class I operations are carried out, the floors must be covered with plastic or metlax sheets and be sloped toward drain traps; the walls are covered with enameled plates or plastic; the ceilings are painted with oil paint. The service personnel are provided with special underclothing, cotton overalls of moleskin or Mylar fabric, and also plastic aprons or oversleeves. Dosimetric monitoring of the contamination of the hands and body, and also a shower bath, are compulsory after operations. /149

All class I operations are carried out in special chambers or cubicles. Operations with isotopes of high activity are performed with the help of mechanical (master-slave) manipulators (Figure 40).

Special ventilation with filters at the outlet for purifying the air must be provided in all areas. An underpressure, with

respect to the third zone, must be produced in the first and second zones in order to prevent leakage of air from the first and second zones, where the probability of contamination is greater than in the third.

When operating with exposed radioactive substances, when the appearance of radioactive aerosols is possible in the area, the service personnel must be provided with respirators for protecting the respiratory system from radioactive substances. Figure 41 shows respirators of the single-use type "Lepestok", and multiple-use type "Astra". FP fabric serves as the filter for trapping radioactive aerosols [41].



/150

Figure 40. Mechanical (master-slave manipulator for operating with isotopes of high activity

Repair operations and decontamination of cubicles and chambers are carried out from the direction of the second zone. When these operations may be accompanied by high air contamination, they are performed in special pressure suits provided with an air supply (Figure 42).

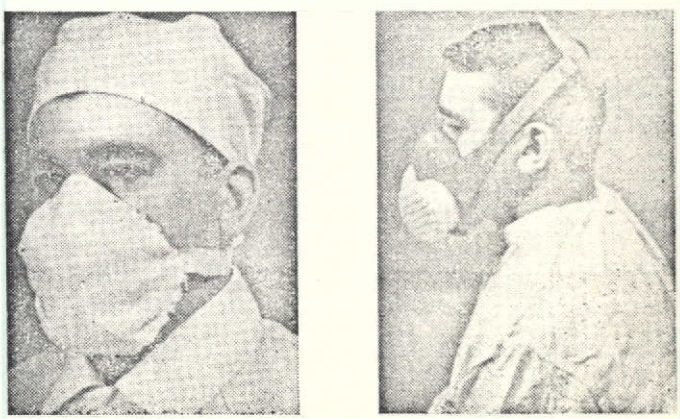


Figure 41. Respirators of the type "Lepestok" (left) and "Astra" (right)

For passages from areas for operations of a higher class into areas for operations of a lower class, it is necessary to monitor the radioactive contamination level by means of

individual shielding (plastic coveralls, aprons, additional boots). After each use, the additional means of individual shielding (film or rubber) must then be subjected, as a rule, to decontamination in the sanitary sluice or another specially drained place.

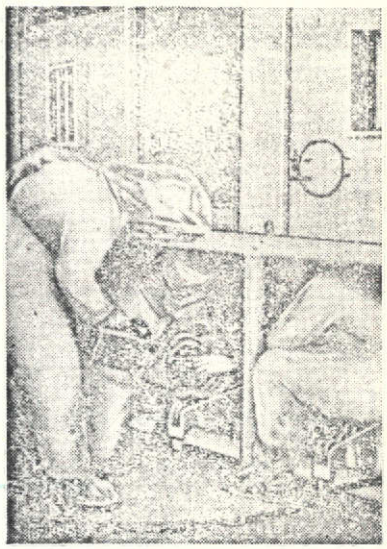


Figure 42. Repair of equipment contaminated with radioactive substances. Workers are dressed in pressure suits

Systematic monitoring of the contamination level of the basic overalls must be carried out along with this. Further use of the overalls is not allowed, according to PSR-72, if their contamination level exceeds the values indicated in Table 15.

The permissible levels of contamination of the external surfaces of gloves, overshoes, and additional means of individual shielding (plastic oversleeves, overalls, coverall, etc.) are the same as the permissible levels of contamination of the surfaces in the working areas in which these means are used.

As for personal clothing and shoes, their contamination by radioactive substances is not permitted. In case of contamination /151
of personal clothing or shoes, they must be immediately directed to decontamination under the control of the radiation safety service, and in case of the impossibility of decontamination, they must be removed and buried as radioactive waste.

Class I and II laboratories must be provided with special containers for the collection of solid and liquid radioactive waste, which are periodically removed to burial points, containers for transfer of radioactive substances within the area and special storage vaults. In some cases, class I laboratories are equipped with special channels for drainage of radioactive waste.

Water faucets in class I and II work areas should be equipped with foot pedals. Conical drainage basins with rounded edges must be fabricated of corrosion-resistant materials or coated with appropriate films; they must also be easily decontaminated.

In order to provide cleaning of the laboratory furniture in case of its contamination, it must be of the simplest design, with smooth surfaces and coated with oil or nitro-enamel point. The furniture is produced with tall legs for convenience in cleaning under it.

It is not necessary to enumerate the other requirements imposed on areas and equipment intended for operation with radioactive substances. They are discussed in detail in PSR-72 [28]. It should be indicated only that the correct organization of operations, systematic disometric monitoring, and spontaneous measures in decontamination when necessary, and also performance of the rules of personal hygiene, are necessary conditions for providing radiation safety which permit eliminating the hazardous effects of radioactive substances on the human organism.

Collection and Disposal of Radioactive Waste

With the increasing development of atomic industry and power. the amount of radioactive substances, which at present cannot be used by man in practical activity, has also increased, i.e., the amount of radioactive waste, which must be treated in some way and buried, has increased. /152

According to its aggregate state, radioactive waste can be liquid, solid, and gas, and depending on the scale of production or use of radioactive substances, their activity can vary over very wide limits: from several mCi to tens and thousands of curies.

Liquid waste is considered radioactive if the content of radioactive substances is greater than the MAPC allowable for water.

Solid waste is considered radioactive if the contamination level of the surface is greater than 500 α-particles/min, or 5000 β-particles/min per area of 100 cm^2; its specific activity (Ci/kg) exceeds the MAPC for water by 100 times, the specific γ-equivalent exceeds 10^{-7} gm-equiv Ra per 1 kg, and the γ-radiation intensity close to the surface is greater than 0.3 mrem/hr [28].

Rough calculations indicate that by 1975, the activity of atomic industrial waste over the whole world will reach 10^{10} Ci, and approximately 10^{12} Ci by the year 2000 [42, 43]. We note for comparison that the activity of SR^{90} accumulated in the atmosphere as a result of nuclear weapons testing is about 10^7 Ci.

Thus, in the total problem of providing radiation safety, the serious questions are those related to the development of those methods of treatment and burial of radioactive waste which would lead to the minimum possible increase of radioactivity of environmental objects and, hence, the effect of radiation on man. With consideration of the expanding scale of the use of atomic energy, works have been intensified in perfecting such methods of treatment and burial of radioactive waste, which have already decreased to a large degree the penetration of radioactive products into the environment.

In order to eliminate the penetration of gaseous radioactive waste into the environment, effective systems for cleaning air of radioactive products have been created at the exhaust of ventilation systems at nuclear reactors, radiochemical industries, in radiochemical laboratories, etc. In accordance with PSR-72 [28], the effectiveness of filters and the height of conduits must be such as to provide a decrease of the contamination of atmospheric air to values not exceeding the MAPC for individuals from the population and the limits of external and internal irradiation doses for this category of the population to the levels specified by NRS-69. /153

On of the main problems in disposing of liquid and solid radioactive waste is the decrease of its volume, which simplifies its isolation from the surrounding medium. For this reason, solid radioactive waste (equipment and instruments contaminated with radioactive substances, rags, overalls, etc.) are compressed, which permits decreasing their volume by 2 - 8 times. For a more reliable immobilization of radioactive substances, all compressed waste is enclosed in cement or asphalt and then buried in special burial grounds. Burial of such blocks at sea at a depth of 500 meters is practiced abroad. However, the question has now been raised about limiting such a method for burying radioactive waste [44].

Burning of combustible solid radioactive waste (rags, overalls, etc.) is practiced at present, which permits decreasing their volume by 40 - 50 times. The remaining radioactive ash is enclosed in cement or asphalt blocks.

The methods of evaporation, chemical precipitation and ion exchange are used to extract radioactive isotopes and to decrease the volume of liquid radioactive waste. The high-activity pulp which is formed is stored in special containers. However, with this method of storage there arise a number of complex problems related to the evolution of heat formed during the radioactive decay process, evaporation monitoring, and trapping gaseous products. Thus, the methods of vitrification and enclosure of radioactive pulp in asphalt or cement have begun to be used in recent years, which provide a more reliable immobilization of radioactivity and permit storage of such blocks in simpler burial grounds [43, 44].

Only in our country has the burial of high-activity pulp underground begun to be accomplished on an industrial scale, which practically completely eliminates the penetration of activity into the environment [44]. This is the most effective general-purpose method for burial of radioactive waste.

If liquid radioactive waste has an activity no more than 10 times greater than the MAPC for water of open reservoirs and their /154

ten-fold dilution with nonradioactive sewage is ensured in the collector of a given institution, then it can be drained in the ordinary industrial channels. If the waste activity is higher than this and its volume exceeds 200 liters/day, then it is necessary to construct special channels with purification installations, where the above-mentioned waste treatment is carried out [28].

It is strictly forbidden to discharge liquid radioactive waste into ponds and lakes intended for cultivation of fish and water fowl, and also into streams and other reservoirs whose water can enter these ponds and lakes.

According to PSR-72, special areas are isolated for burial of radioactive waste, which are located beyond the limits of the prospective development of populated areas, rest areas, sanitaria, etc., and no closer than 500 meters from open reservoirs. A sanitary-protective zone is created around the burial area.

It should be noted that from the moment of the creation of atomic industry, the requirements imposed on methods for the collection and burial of radioactive waste have been stricter than in other branches of industry. As a result, in spite of the enormous amount of radioactive waste accumulated up to the present time, the radiation loads on the population caused by waste is so small that it is difficult to measure. This indicates that existing methods of treatment and burial of radioactive waste are sufficiently effective. However, taking into account the prospects for the use of atomic energy, it is necessary to improve the waste treatment system to efficiently eliminate penetration of radioactive substances into the environment.

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